Appendix 6A EVALUATION OF UNDAMAGED FUEL UNDER ACCIDENT ACCELERATIONS

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APPENDIX 6B

DAMAGED FUEL CLADDING STRUCTURAL EVALUATION

6B.1 Introduction

The purpose of this appendix is to demonstrate structural integrity of the damaged fuel cladding in the TN-68 basket following normal and off-normal loading conditions of storage and onsite transfer (required for Part 72 License) and normal condition of offsite transport (required for Part 71 License: included here for information only).

Note: Although this appendix discusses low burnup and high burnup scenarios, only damaged fuel assemblies are limited to maximum bundle average burnup of ≤ 45 GWd/MTU.

In this appendix, the damaged fuel is defined as fuel assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks and pinhole leaks as defined in Technical Specification Section 2.1.1. Damaged fuel must be capable of being handled by normal means in order to be stored in the TN-68.

This appendix evaluates stresses in the damaged fuel cladding associated with normal and off-normal conditions of on-site transfer/storage and off-site transport. It also presents a fracture mechanics assessment of the cladding using conservative assumptions regarding defect size geometry and amount of oxidation in the cladding material. These evaluations demonstrate the structural integrity of the damaged fuel cladding under normal and off-normal conditions.

The TN-68 cask and fuel basket is designed to store 68 intact fuel assemblies, or no more than 8 damaged and the remainder intact, for a total of 68 standard BWR fuel assemblies per canister. All the fuel assemblies, intact or damaged, consist of BWR fuel assemblies with Zircaloy cladding. Damaged fuel assemblies may only be stored in eight peripheral compartments of the TN-68 fuel basket fitted with end caps to retain and retrieve damaged fuel fragments.

6B.2 Design Input / Data

The design inputs are summarized in the following table. The design inputs and assumptions are the same as described in Appendix 6A.

Design Parameters of BWR Fuel Assemblies

Tube Arrays	**7 x 7**	**8 x 8**	**8 x 8**	**8 x 8**	**8 x 8**	**9 x 9**	**10 x 10**
GE Designation	GE2, GE3	GE4	GE5	GE8	GE9, GE10	GE11, GE13	GE12
No. of Full Length Fuel Rods	49	63	62	60	60	66	78
Maximum Active Fuel Length (in.)	144	146	150	150	150	146	150
Fuel Tube OD (in.)	0.563	0.493	0.483	0.483	0.483	0.44	0.404
Corroded Fuel Tube OD (in.)	0.5576	0.4876	0.4776	0.4776	0.4776	0.4346	0.3986
Clad Thickness (in.)	0.032	0.034	0.032	0.032	0.032	0.028	0.026
Corroded Clad Thickness (in.)	0.0293	0.0313	0.0293	0.0293	0.0293	0.0253	0.0233
Fuel Tube I.D. (in.)	0.499	0.425	0.419	0.419	0.419	0.384	0.352
Fuel Tube Radius, mid-thickness (in.)	0.2642	0.2282	0.2242	0.2242	0.2242	0.2047	0.1877
Corroded Fuel Tube Area (in^2)	0.0486	0.0449	0.0413	0.0413	0.0413	0.0325	0.0275
Corroded. Fuel Tube M.I. (in^4)	1.702×10^{-3}	1.173×10^{-3}	1.041×10^{-3}	1.041×10^{-3}	1.041×10^{-3}	0.684×10^{-3}	0.486×10^{-3}
Irradiated Yield Stress at 750 °F(1) (psi)	69,000	69,000	69,000	69,000	69,000	69,000	69,000
Young's Modulus E at 750°F (1) (psi)	1.21E+07	1.21E+07	1.21E+07	1.21E+07	1.21E+07	1.21E+07	1.21E+07

Note:
1. Values are calculated from PNNL report [36] with very small strain input.

6B.3 Loads

6B.3.a Part 72 Normal and Off-normal Condition Loads

The damaged fuel ($\leq$ 45 GWd/MTU burnup) inside the TN-68 fuel basket is subjected to following normal and off normal condition Part 72 loads:

- Dead Weight
- Internal Pressure
- Thermal
- Transfer Load (Inertia Loads associated with moving the TN-68 cask in vertical position from the fuel loading area to the ISFSI site), which consists of 1*g* in the longitudinal, 1*g* in the transverse and 1*g* in the vertical direction.

The stresses due to the dead weight are insignificant. No internal pressure is assumed for the damaged fuel. The cladding is assumed to be able to expand due to thermal loads and thus no thermal-induced stresses are considered. However, the temperature of the cladding is considered for selection of allowable stresses. Therefore, the structural integrity of the damaged fuel ($\leq$ 45 GWd/MTU burnup) is evaluated in this section only for the following transfer/storage loads.

1g Vertical Loads

The maximum g load acting on the damaged fuel rods subjected to 1g vertical load is conservatively taken as 2g (1g deadweight + 1g external load). The damaged fuel rod structural integrity under this 2g load is assessed by computing the compressive stress in the cladding by ratioing the stresses from the 15g end drop (2/15) given Section 6B.5. The results of the compressive stress are shown in the following table.

<u>**Axial Compressive Stresses due to Vertical Loads**</u>

Tube Arrays	**7 x 7**	**8 x 8**	**8 x 8**	**8 x 8**	**8 x 8**	**9 x 9**	**10 x 10**
GE Designation	GE2, GE3	GE4	GE5	GE8	GE9, GE10	GE11, GE13	GE12
No. of Full Length Fuel Rods	49	63	62	60	60	66	78
Fuel Assembly Weight (lb)	705	705	705	705	705	705	705
Sectional Area (in^2)	0.0486	0.0449	0.0413	0.0413	0.0413	0.0325	0.0275
Compressive Stress (ksi)	0.59	0.50	0.55	0.57	0.57	0.66	0.66

The axial stresses in the fuel rod are compressive stresses and are significantly less than the irradiated yield stress of the cladding material (69.0 ksi, at 750°F, data is calculated from PNNL report [36] with very small strain rate input). Therefore, the fuel rods will maintain their structural integrity when subjected to the 2g applied load in the vertical direction.

<u>1g Longitudinal/Transverse Load</u>

The maximum g load acting on the damaged fuel rods under longitudinal/transverse load is conservatively taken as 2g lateral load. The damaged fuel rod structural integrity under this load is assessed by computing the bending stress in the cladding by ratioing the stresses from the 35g side drop (2/35) given Section 6B.6.(1) The results of the maximum bending stress are shown in the following table.

Note 1: The maximum bending stresses in the 35g side drop analysis is due to overhang (top portion of the fuel rods). The stress outside the overhang location is much lower. During the transfer condition the cask is vertical, the fuel rod is supported at the bottom of the cask, and there will be no overhang of the fuel rod at the top of the basket. The 35g side drop analysis is nonlinear (Contact Element), most of the nonlinearity occurred at the overhang location, and the center and bottom end of the fuel rod remain mostly linear. Since the stress at the center and bottom portion is lower than the overhang location, ratioing the stress from the overhang location is conservative.

Summary of Stress Results for the Longitudinal/Transverse Load

	7x7	8x8	8x8	8x8	8x8	9x9	10x10
GE Designation	GE2, GE3	GE4	GE5	GE8	GE9, GE10	GE11, GE13	GE12
Fuel Cladding O.D. (in.)	0.5576	0.4876	0.4776	0.4776	0.4776	0.4346	0.3986
Fuel Cladding I.D. (in.)	0.499	0.425	0.419	0.419	0.419	0.384	0.352
Fuel Cladding thickness (in.)	0.0293	0.0313	0.0293	0.0293	0.0293	0.0253	0.0233
Max Bending Stress, S_b (ksi)	2.31	2.39	2.75	2.75	2.75	2.33	2.50

The maximum bending stresses in the fuel rod are significantly less than the irradiated yield stress of the cladding material (69.0 ksi).

The same fracture mechanics evaluation methodology described in Section 6B.8 is also used to evaluate the maximum cladding bending stresses due to the 2g lateral load case. The fuel cladding material fracture toughness (K_{IC}) is taken from the lower-bound value presented in reference [13] as described below.

1. Through-wall Axial Crack in the Cladding

As described in Section 6B.8.d, the maximum stress intensity factor results indicate that any existing axial through-wall crack in the spent fuel cladding would not sustain further damage from any additional load. From the linear elastic fracture mechanics point of view, the axial through-wall crack will not cause fracture.

2. Through-wall Circumferential Crack in the Cladding Under Bending

Section 6B.8.c evaluates this crack model using the model shown in Figure 6B-11. The calculation was performed using the computer code **pc-CRACK** [34]. Bending stresses at 1 ksi, 5 ksi, and 10 ksi to 80 ksi at 10 ksi intervals were calculated for a parametric fracture mechanics evaluation.

Figure 6B-13 presents the applied stress intensity factor versus half crack length for GE fuel cladding. The stress intensity factors (K) were presented for several levels of applied bending stress, from 1 ksi to 80 ksi.

Using the results in Figures 6B-13, for a given applied stress level along with selected through-wall circumferential crack length, the applied K value can be determined.

The maximum calculated bending stress from table above is 2.75 ksi for GE 8x8; conservatively using a stress level of 5 ksi and a half crack length of 0.05 in. (this is equivalent to crack length of 0.1 in., reference [26] indicates that typical crack length is less than 0.039 in.), the applied K

value from Figure 6B-13 is determined to be approximately 2.0 ksi-in$^{1/2}$.

This calculated stress intensity factor (K) is compared with experimentally obtained fracture toughness, K_{IC}, of irradiated Zircaloy cladding material. Reference [13, pg. 4-1] suggests a typical lower-bound value of K_{IC} (PWR or BWR) for end-of-life burnup at 20°C (68°F) with relatively high hydrogen concentration (≈ 750 ppm) is in the range of 18-20 MPa m$^{1/2}$ (16.36-18.18 ksi in$^{1/2}$). Therefore, a K_{IC} value of 16.36 ksi in$^{1/2}$ is used for the fracture evaluation. The K_{IC} = 16.36 ksi in$^{1/2}$ fracture toughness value is considered conservative since it is measured at relatively low temperatures. Also, the stress intensity ratio is on the order of 0.12 (2/16.36) which translates into a factor of safety on the order of 8 which accounts for any unknown effects.

This evaluation demonstrates that the damaged fuel assemblies (≤ 45 GWd/MTU burnup) in the TN-68 cask will retain their structural integrity when subjected to normal and off-normal condition transfer and storage loads. Therefore, the retrievability of the damaged fuel assemblies is assured when subjected to any of these normal and off-normal transfer and storage loads.

6B.3.b Part 71 Normal Condition Loads

The evaluations of the 1 foot end drop and 1 foot side drop are for information only for Part 72 application. The structural integrity of the damaged fuel cladding due to 1 foot end drop and 1 foot side drop will be addressed and analyzed in the future Part 71 application.

The damaged fuel is evaluated for the following normal condition 10CFR Part 71 off-site transportation loads:

- 1 foot end and side drop loads
- Vibratory loads
- Shock load
- Lifting and Tie-down loads

During one-foot end and side drops, fuel assemblies are subjected to 15g and 35g loads respectively [10].

Vibratory loads of 0.30g in longitudinal direction, 0.30g in the transverse direction and 0.60g in the vertical direction, taken from Reference [4] are considered representative for a truck loaded cask. The vibration load of 0.19g in the longitudinal direction, 0.19g in the transverse direction and 0.37g in the vertical direction, taken from Reference [4], are considered representative for a rail car loaded cask [5].

The shock load of 4.7g in the longitudinal and 4.7g in the lateral and vertical directions for a rail car loaded cask (bounding values between rail and truck transport) during off-site transport are also taken from Reference [5].

Lifting load of 6g vertical is taken fromPart71-45(a). Tie-down loads 2g (vertical)/5g (lateral)/10g (longitudinal) are taken from Part71-45(b).

All of the above loads however are bounded by 1 foot end drop (15*g*) and 1 foot side drop (35*g*) transport load. Therefore, structural integrity of the damaged fuel for the normal conditions of Part 71 is evaluated only for the one-foot end and side drop conditions.

6B.4 Evaluation Criteria

The retrievability of the damaged fuel in the TN-68 Cask is assured if the damaged fuel cladding retains its structural integrity when subjected to normal and off normal loads. Per the damaged fuel definition in Section 6B.1, the damaged fuel rods loaded in the TN-68 basket may have cladding defects greater than hairline cracks or pinhole leaks. However, under normal and off-normal loads, the original defects (such as cracks or pinholes or missing grid) should not change significantly so that the damaged fuel can be retrieved.

The damaged fuel cladding needs to meet the following criteria to ensure their structural integrity and thus be retrievable:

- Fuel cladding stresses under normal and off-normal load conditions are less than the irradiated yield strength of the cladding material.
- Stability of the cladding tube is maintained (i.e., no buckling occurs).
- The stress intensity factor, K_I, of the fuel cladding tube geometry considering through-wall flaw is less than experimentally determined fracture toughness, K_{Ic}, considering temperature and irradiation effects.

6B.5 One Foot End Drop (15g) Damaged Fuel Evaluation

During off site transport (Part 71) the damaged fuel assemblies need to be evaluated for one foot end drop. The maximum *g* load acting on the damaged fuel rod subjected to one foot end drop of the TN-68 cask is 15g.

Damaged Fuel One Foot End Drop Stress Evaluation

The maximum g load acting on the damaged fuel rod subjected to one foot end drop is 15g. The calculation assumes that no credit is taken for the fuel pellet, i.e., the loads are entirely taken by the cladding.

Stress Analysis of GE2-7x 7 Fuel Assemblies

Number of rods per assembly = 49
Therefore, force per rod = (705*15)/49 = 215.82 lb
Area of the cladding = 0.0486 in^2
Axial compressive stress in the rod = 215.82/0.0486 = 4,440 psi = 4.44 ksi

Using the same methodology, axial compressive stresses for the cladding of all assembly types are calculated and summarized in the following table.

Axial Compressive Stresses due to End Drop (15g)

Tube Arrays	7 x 7	8 x 8	8 x 8	8 x 8	8 x 8	9 x 9	10 x 10
GE Designation	GE2, GE3	GE4	GE5	GE8	GE9, GE10	GE11, GE13	GE12
No. of Full Length Fuel Rods	49	63	62	60	60	66	78
Fuel Assembly Weight (lb)	705	705	705	705	705	705	705
Sectional Area (in^2)	0.0486	0.0449	0.0413	0.0413	0.0413	0.0325	0.0275
Compressive Stress (ksi)	4.44	3.74	4.13	4.27	4.27	4.93	4.94

The axial stresses in the fuel rod are compressive stresses and are significantly less than the irradiated yield stress of the cladding material = 73,712 psi (750°F). The maximum calculated cladding temperature is 622°F (Table 4.6-2). Therefore, the fuel rods will maintain their structural integrity when subjected to the one foot end drop load. Also, 15g axial stresses are significantly lower than 35g one foot side drop stresses and are enveloped by the side drop load fracture toughness evaluation in Section 6B.8.

6B.6 One Foot Side Drop Damaged Fuel Evaluation

The maximum *g* load acting on the damaged fuel rods under one foot side drop load is 35*g*. The damaged fuel rod structural integrity under one foot side drop load is assessed by computing the bending stress in the rod and comparing it with the yield stress of the cladding material.

The ANSYS models used for 75g side drop analyses as described in Appendix 6A are used for the one foot 35g side drop analyses. The boundary conditions, material properties, and assumptions are the same as described in Appendix 6A. The model is subjected to loads due to cladding tube mass, fuel pellets mass, and the fuel assembly end fitting mass. However, no credit is taken for fuel pellet moment of inertia. The loads are entirely taken by the cladding. The following table summarizes the stress results for the 35g side drop analyses. The results, when compared to those in Table 6A-4, show that bending stresses at 35g are lower than at 75g, but they do not decrease linearly. At the lower g-load there is less contact among the numbers of fuel rods at the top edge of the basket which distributes the loads on fewer rods, thus increasing the stress at the bottom rod cladding. The calculated bending stresses are less than the irradiated yield stress of the cladding material, 73,712 psi (750° F).

Summary of Stress Results for 35g Side Drop

	7x7	**8x8**	**8x8**	**8x8**	**8x8**	**9x9**	**10x10**
GE Designation	GE2, GE3	GE4	GE5	GE8	GE9, GE10	GE11, GE13	GE12
Fuel Cladding O.D. (in.)	0.5576	0.4876	0.4776	0.4776	0.4776	0.4346	0.3986
Fuel Cladding I.D. (in.)	0.499	0.425	0.419	0.419	0.419	0.384	0.352
Fuel Cladding thickness (in.)	0.0293	0.0313	0.0293	0.0293	0.0293	0.0253	0.0233
Max Bending Stress, S_b (psi)	40,442	41,753	48,172	48,172	48,172	40,715	43,823

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6B.9 Conclusions

The maximum computed stresses due to end and side drops in the fuel rods and their ratios to the irradiated yield stress of the cladding material are summarized in Table 6B-1. From this table, it can be concluded that stresses for all load cases considered are significantly less than the yield stress of the Zircaloy cladding material.

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Table 6B-1

Maximum Computed Fuel Rod Stresses and their Ratio to Yield Strength

Load	Maximum Computed Stress (ksi)				Zircaloy Cladding Yield Strength (at 750°F) (ksi)	Ratio of Maximum Computed Stress to Yield Strength
	(7×7) Fuel	(8×8) Fuel	(9×9) Fuel	(10×10) Fuel		
1-foot End Drop	4.44	4.27	4.93	4.94	73.71	0.07
1-foot Side Drop	40.44	48.17	40.72	43.82	73.71	0.65

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CHAPTER 7

CONFINEMENT

7.1 Confinement Boundary

The confinement boundary consists of the inner shell and bottom plate, shell flange, lid outer plate, lid bolts, penetration cover plate and bolts and the inner metallic O-rings of the lid seal and the two lid penetrations (vent and drain). The confinement boundary is shown in Figure 1.2-1. The construction of the confinement boundary is shown on drawings 972-70-1, 2 and 3 provided in Section 1.5. The confinement vessel prevents leakage of radioactive material from the cask cavity. It also maintains an inert atmosphere (helium) in the cask cavity. Helium assists in heat removal and provides a non-reactive environment to protect fuel assemblies against fuel cladding degradation which might otherwise lead to gross rupture.

7.1.1 Confinement Vessel

The TN-68 confinement vessel consists of: an inner shell which is a welded, carbon steel cylinder with an integrally-welded, carbon steel bottom closure; a welded flange forging; a flange and bolted carbon steel lid with bolts; and vent and drain covers with bolts. The overall confinement vessel length is 189.0 in. with a wall thickness of 1.5 in. The cylindrical cask cavity has a diameter of 69.5 in. and a length of 178 in.

The confinement shell and bottom closure materials are SA-203 Grade E and the shell flange is SA-350 Grade LF3. The confinement lid material is SA-203 Grade E or SA-350 Grade LF3.

The cask design, fabrication and testing are performed under Transnuclear's Quality Assurance Program which conforms to the criteria in Subpart G of 10CFR72.

The materials of construction meet the requirements of Section III, Subsection NB-2000 and Section II, Material specifications or the corresponding ASTM Specifications. The materials used in the confinement boundary conform to the requirements of NB-2121 and NB-2130. The confinement vessel is designed to the ASME Code, Section III, Subsection NB, Article 3200. The confinement vessel is fabricated and examined in accordance with NB-2500, NB-4000 and NB-5000. Welding materials used in confinement welds or welds to the confinement components conform to the requirements of NB-2400 and to the material specification requirements of Section II, Part C of the ASME B&PV Code.

The confinement vessel is hydrostatically tested in accordance with the requirements of the ASME B&PV Code, Section III, Article NB-6200 with the exception that the confinement vessel is installed in the gamma shield shell during testing. The confinement vessel is supported by the gamma shield during all design and accident events.

Even though the code is not strictly applicable to storage casks, it is the intent to follow Section III, Subsection NB of the Code as closely as possible for design and construction of the confinement vessel. The casks may, however, be fabricated by other than N-stamp holders and

materials may be supplied by other than ASME Certificate Holders. Thus the requirements of NCA are not imposed. TN's quality assurance requirements, which are based on 10CFR72 Subpart G and NQA-1 are imposed in lieu of the requirements of NCA-3850. This SAR is prepared in place of the ASME design and stress reports. Surveillances are performed by TN and utility personnel rather than by an Authorized Nuclear Inspector (ANI).

The weld of the bottom inner plate to the confinement shell is a Category C, Type 2 corner weld in accordance with Figure NB-4243-1 of the ASME Code. In accordance with NB-5231, Type 2 Category C full penetration corner welded joints require the fusion zone and the parent metal beneath the attachment surface to be ultrasonically examined after welding. If this weld is performed on the confinement vessel after assembly with the outer shell, the UT inspection will be performed on a best efforts basis. It may not be possible to do a complete UT inspection, since the outer diameter of the shell is inaccessible. The joint will be examined by the radiographic method and either the liquid penetrant or magnetic particle methods in accordance with the ASME Code Subsection NB.

Paragraph NB-4213 requires the rolling process used to form the inner vessel be qualified to determine that the required impact properties of NB-2300 are met after straining by taking test specimens from three different heats. If the plates are made from less than three heats, each heat will be tested to verify the impact properties.

7.1.2 Confinement Penetrations

There are two penetrations through the confinement vessel, both in the lid. One is the drain port and the other is the vent port. A double O-ring seal mechanical closure is provided for each penetration. Each penetration contains a quick disconnect coupling for ease of operation.

7.1.3 Seals and Welds

The confinement boundary welds consist of the circumferential welds attaching the bottom closure and the top flange to the vessel shell. Also, the longitudinal weld(s) on the rolled plate, closing the cylindrical vessel shell, and the circumferential weld(s) attaching the rolled shells together are confinement welds.

Double metallic seals are utilized on the lid and the two lid penetrations. Helicoflex HND or equivalent seals may be used. The seals are shown in Figure 7.1-3. The internal spring and lining maintain the necessary rigidity and sealing force, and provide some elastic recovery capability. The outer aluminum jacket provides a ductile material against the sealing surfaces. The jacket also provides a connecting sheet between the inner outer seals. Holes in this sheet allow for attachment screws and for communication between the overpressure system and the space between the seals. This sheet, which is about 0.020 inch thick, has insufficient strength to transmit radial forces great enough to overcome the axial compressive forces on the seals, which are over 1000 lb/inch of seal length. Additional information on the seals is provided in Section 2.3.2. The overpressure port seal is a single metallic seal of the same design, Helicoflex HN200 or equivalent.

All TN-68 surfaces which mate with the metallic seals are stainless steel.

The use of a double seal system allows the TN-68 cask to have a pressure monitoring of the interspace between the seals (See Section 2.3.2). This combined cover-seal pressure monitoring system always meets or exceeds the requirement of a double barrier closure which guarantees tight, permanent confinement. When the cask is placed in storage, a pressure greater than that of the cavity is set up in the gaps (interspace) between the double metallic seals of the lid and the lid penetrations. A decrease in the pressure of the monitoring system would be signaled by a pressure transducer/switch in the overpressure system.

The lid and penetration seals described above are contained in grooves. A high level of sealing over the storage period is assured by utilizing seals in a deformation-controlled design. The deformation of the seals is constant since bolt loads assure that the mating surfaces remain in contact. The seal deformation is set by the original diameter and the depth of the groove.

The nominal diameter of the lid seal is 6.6 mm, and the nominal groove depth is 5.6 mm. At 1 mm compression, the sealing force is 245 N/mm (1399 lb/inch)[11]. The total force of the double seal is 633,800 lb. The total preload of the 48 lid bolts is 2,897,000 lb, which is greater than the combined force of the seals and internal pressure, 1,141,000 lb (Section 3A.3).

The nominal diameter of the port seals is 4.1 mm, and the nominal groove depth is 3.2 mm. At 0.9 mm compression, the sealing force is 200 N/mm (1142 lb/inch). The total force of the double seal is 37,900 lb. The total preload of the 8 cover bolts is 63,700 lb, which is greater than the combined force of the seals and internal pressure, 40,000 lb.

The sealing force is maintained by the seal's internal spring. Due to creep, the sealing pressure decreases with increasing temperature as shown in the following table[11]. The ratios P_T/P_{20} compare the seal pressure at temperature T °C to the seal pressure at 20 °C. The long-term temperature limit is the point at which the sealing pressure becomes zero due to creep (P_{Tmax}=0). The maximum normal temperature experienced by the seals in the TN-68 is 212 °F (Table 4.3-1), below the 119 °C evaluated in the following table.

Seal	$P_{119\,C}/P_{20\,C}$ (119 °C = 247 °F)	$P_{200\,C}/P_{20\,C}$ (200 °C = 392 °F)	Temperature limit T_{max}
Lid, 6.6 mm	(439/670) = 66%	(250/670) = 37%	340 °C (644 °F)
Ports, 4 mm	(364/600) = 61%	(170/600) = 28%	280 °C (536 °F)

$P_{20\,C}$ and $P_{200\,C}$ from Reference 11; $P_{119\,C}$ by linear interpolation; sealing pressure P in N/mm^2 (referred to as "Intrinsic Power Pu" in reference 11)

The maximum radial force on the seals is from the 6.0 atm abs overpressure system. Using the compressed seal height of 5.6 mm, this results in a force per unit seal length of about

5.0 atm gage*14.7 psi/atm*(5.6/25.4)inch = 16 lb/inch

which is negligible compared to the compressive (axial) forces of over 1000 lb/inch. Because the maximum pressure is between the two seals, the direction of this force is such that the seals are supported by the walls of the seal groove. However, the seals are designed to retain pressure in either direction.

Helicoflex metallic seals are all capable of limiting leak rates to less than 1 x 10^{-7} ref cm^3/s. After loading, all lid and cover seals are leak tested in accordance with ANSI N14.5. The acceptable total cask leakage (both inner and outer seals combined) is 1 x 10^{-5} ref cm^3/s.

7.1.4 Closure

The confinement vessel contains an integrally-welded bottom closure and a bolted and flanged top closure (lid). The flanged lid plate is attached to the cask body with 48 bolts. The bolt torque required to seal the metallic seals located in the lid and maintain confinement under normal and accident conditions is provided in Drawing 972-70-1. The closure bolt analysis is presented in Appendix 3A.3.

As previously mentioned, the lid contains two penetrations which are sealed by flanged covers fastened to the lid by 8 bolts each. The bolt torque required to seal the metallic seals in the penetration covers and maintain confinement under normal and accident conditions is provided in Drawing 972-70-1.

7.1.5 Monitoring of System Confinement

An overpressure monitoring system is part of the TN-68 design. The pressure in the monitoring system is greater than that of the cask cavity and the cask cavity pressure is greater than ambient. In this configuration, neither in-leakage of air nor out-leakage of cavity gas is possible.

If a leak existed in the seals, the design of the TN-68 overpressure system is such that the leak will either be to the atmosphere or to the cask cavity. Leakage from the cask cavity past the higher pressure of the overpressure system is physically impossible.

The seals are collectively leak tested to 1 x 10^{-5} ref cm^3/s. Using the methodology of ANSI N14.5[(2)], an equivalent maximum hole size is estimated based upon test conditions of equivalent air leaking from 1 atm abs to 0.01 atm abs in ambient temperature conditions (77°F or 25°C) and the maximum acceptable leak of 1 x 10^{-5} ref cm^3/s. The leakage hole length is assumed to be the same as the metal seal width, 0.5 cm. The equivalent maximum hole size is calculated below.

$$L_u = (F_c + F_m)(P_u - P_d)(P_a/P_u) \text{ cc/sec at } T_u, P_u$$

Other definitions:

L_u	= upstream volumetric leakage rate, cc/sec = 1 x 10^{-5} ref cm^3/s (Test Leak Rate)
F_c	= coefficient of continuum flow conductance per unit pressure, cc/atm-sec
F_m	= coefficient of free molecular flow conductance per unit pressure, cc/atm-sec
P_u	= fluid upstream pressure, atm abs = 1.0 atm abs
P_d	= fluid downstream pressure, atm abs = 0.01 atm abs
D	= leakage hole diameter, cm

a	= leakage hole length, cm = 0.5 cm (assuming leak path length is on the order of the metal seal width)
μ	= fluid viscosity, cP = 0.0185 cP (from ANSI N14.5, Table B.1)
T	= fluid absolute temperature, °K = 298°K
M	= molecular weight, g/mol = 29.0 g/mol (from ANSI N14.5, Table B.1)
P_a	= average stream pressure = ½ (P_u + P_d), atm abs = 0.505 atm abs

$L_u = (F_c + F_m)(P_u - P_d)(P_a/P_u)$ cc/sec

where:

$F_c = (2.49\text{x}10^6 \times D^4)/(a\mu)$ cc/atm-sec

$F_m = \{3.81\text{x}10^3 \times D^3 \times (T/M)^{0.5}\} / \{aP_a)$ cc/atm-sec

Substituting:

$F_c = (2.49\text{x}10^6 \times D^4)/(0.5 \times 0.0185) = 2.69 \text{ x } 10^8 D^4$

$F_m = \{3.81\text{x}10^3 \times D^3 \times (298/29.0)^{0.5}\} / \{0.5 \times 0.505) = 4.84 \text{ x } 10^4 D^3$

$L_u = (F_c + F_m)(P_u - P_d)(P_a/P_u)$ cc/sec

$1 \text{ x } 10^{-5} = (F_c + F_m)(1.0 - 0.01)(0.505 / 1.0)$

$F_c + F_m = 2 \text{ x } 10^{-5}$

Solving the simultaneous equations, the equivalent hole diameter, D, is $4.825 \text{ x } 10^{-4}$ cm.

During operations, the overpressure system is initially back filled with 6 atm abs (73.5 psig) of Helium at standard temperature. The temperature of the helium in the O.P. tank at equilibrium is assumed to be 174°F (79°C)*. The pressure in the overpressure system at this temperature will be 7.09 atm abs (89 psig). Assuming the overpressure system is leaking to the atmosphere, the leak rate is defined using the equations of ANSI N14.5:

$L_{u,He} = (F_c + F_m)(P_u - P_d)(P_a/P_u)$ cc/sec

$F_c = (2.49\text{x}10^6 \times D^4)/(a\mu)$ cc/atm-sec

$F_m = \{3.81\text{x}10^3 \times D^3 \times (T/M)^{0.5}\} / \{aP_a)$ cc/atm-sec

Where:

$L_{u,He}$	= helium volumetric leakage rate
P_u	= 7.09 atm abs
P_d	= 1.0 atm abs
D	= $4.825 \text{ x } 10^{-4}$ cm
a	= 0.5 cm
μ	= 0.0223 cP (for helium at 352 K)

* The assumed OP temperature is based on thermal analysis for 21.2 kW heat load. Since the axial heat transfer in the TN-68 cask from the basket to the OP system is not significant, the OP system temperature at thermal equilibrium for 30 kW heat load would not differ much from the value assumed for the 21.2 kW case.

$T = 352°K$
$M = 4.0$ g/mol
$P_a = ½ (P_u + P_d) = 4.04$ atm abs

Substituting:

$F_c = \{2.49E+06 \times (4.825E-04)^4\}/(0.5 \times 0.0223) = 1.21E-05$
$F_m = \{3.81E+03 \times (4.825E-04)^3 \times (352/4)^{0.5}\} / (0.5 \times 4.04) = 1.99E-06$

$L_{u,He} = (F_c + F_m)(P_u - P_d)(P_a/P_u)$
$L_{u,He} = (1.21E-05 + 1.99E-06)(7.09 - 1.0)(4.04/7.09)$
$L_{u,He} = 4.9E-05$ cc/sec of Helium

Over the first year, the maximum volume leaked from the overpressure system is:

$V = 4.9E-05$ cc/sec $\times$ (365 days/year $\times$ 24 hrs/day $\times$ 3600 sec/hr) = 1544 cc at T_u, P_u

The OP system tank basically consists of a 6" diameter schedule 80 pipe (27" long) and two 6" diameter schedule 80 end caps. The volume of the tank is 835 in^3. The volume of the OP system is increased to 900 in^3 (14750 cc) to include the OP system tubing and the space between the metallic seals in the lid and penetrations. Corresponding, the pressure is reduced by the following in the first year:

$P_{OP\ released} = P_{OP\ Sys,\ Initial} \times \{V_{released} / V_{OP\ Sys}\}$
$P_{OP\ released} = 6.53$ atm (1544cc / 14750cc) = 0.74 atm

The overpressure system pressure is also corrected for the corresponding drop in temperature over the first year. At the end of the first year, the overpressure system pressure is 6.34 atm abs (78.5 psig). These calculations are repeated every year for the 20 year life of the cask. Figure 7.1-1 illustrates the pressure drop from the overpressure system to the atmosphere. Figure 7.1-1 also illustrates the pressure drop in the cask cavity due to fuel cooling.

If a leak is to the cask cavity rather than the atmosphere, the pressure drop in the overpressure system is calculated using a downstream pressure of 2.2 atm abs (17.64 psig). Figure 7.1-1 also illustrates the results of this analysis. In this scenario, the corresponding increase in the cask cavity pressure is negligible.

As shown above, the monitoring system pressure is greater than the cask cavity or atmospheric pressure assuming a leak based on the conservative initial acceptance test leak rate of 1×10^{-5} ref cm^3 /s. Typically, helicoflex metallic seals result in joints with much lower leak rates than the acceptance criteria. Therefore, no leakage will occur from the cask cavity during the storage period.

The pressure in the overpressure system will be monitored over the lifetime of the cask. To allow time to diagnose and correct any problems, the overpressure monitoring system is set to alarm if the overpressure system drops below 3.0 atm abs (29.4 psig). This alarm setpoint

ensures that pressure decreases in the overpressure monitoring system are identified well before any potential out leakage from the cask cavity occurs.

7.2 Requirements for Normal Conditions of Storage

7.2.1 Release of Radioactive Material

The TN-68 dry storage cask is designed to provide storage of spent fuel for at least 40 years. The cask cavity pressure is always above ambient during the storage period as a precaution against the in-leakage of air which might be harmful to the fuel. Since the confinement vessel consists of a steel cylinder with an integrally-welded bottom closure, the cavity gas can escape only through the lid closure system. In order to ensure cask leak tightness, two systems are employed. First, all bolted closures are provided with double seals. Second, the interspace between the seals is pressurized to provide a positive pressure gradient. If the inner seals were to leak, helium would flow into the cask cavity and radioactive material would not be released. If the outer seals were to leak, helium would leak from the overpressure system to the exterior, and no radioactive material would be released.

The cask loadings for normal conditions of storage are given in Section 2.2.5. It is shown that the seals are not disturbed by any of the loadings and thus, the cask confinement is maintained.

7.2.2 Pressurization of Confinement Vessel

The TN-68 cask cavity's equilibrium pressure during normal storage conditions with no fuel rod rupture is 2.2 atm abs (17.6 psig). The internal pressure is determined on the basis that a minimum of 1 atm pressure must exist on the coldest day at the end of life. Pressure variations due to daily and seasonal changes in ambient temperature conditions will be small due to the large thermal capacity of the cask. The initial pressure of 2.2 atm abs assures that at the end of 40 years, on the coldest day (–20°F ambient), the internal pressure of the cask is:

$$P_{cavity} = 2.20 \text{ atm abs} \times (596°R / 862°R) = 1.5 \text{ atm abs } (7.7 \text{ psig})^{\dagger}$$

Therefore, the internal pressure of the cask is above the 1 atm minimum.

7.2.2.1 Pressure at Helium Backfill

The maximum normal operating pressure is 2.2 atm abs. A steady state run of the full cask model described in Section 4.5.1 determines the average cavity gas temperature after completion of the helium backfilling. An ambient temperature of 70°F is considered for this run. The average gas cavity temperature is 350°F (810°R), and is retrieved from the model using the methodology described in Section 4.7.1.

Section 4.7.1 shows, once the cask reaches thermal equilibrium, the average cavity gas temperature with 100°F ambient air and maximum solar load is 402°F (862°R). Therefore, the pressure at helium backfill should be equal to or less than:

$$2.2 \text{ atm abs } (810°R / 862°R) = 2.07 \text{ atm abs } (15.7 \text{ psig})$$

† 596°R is the average gas cavity temperature after 40 years of storage assuming an external ambient temperature of -20°F and 21.2 kW initial heat load.

7.2.2.2 Pressure under 100°F Ambient Air Temperature, Maximum Insolation, 10% Fuel Rod Failure

See Section 4.7 for calculation of the cask cavity pressure under normal, off-normal, and accident conditions.

7.3 Confinement Requirements for Hypothetical Accident Conditions

7.3.1 Source Terms for Confinement Calculations

Section 5.2.4 provides the definitions and source terms for three combinations of burnup, enrichment, and cooling time for 8x8 fuel: design basis (DBF-68), medium burnup (MBF-68) and high burnup (HBF-68). These represent bounding combinations of fuel characteristics allowed for storage under the fuel qualification flowchart in TN-68 Technical Specification Figure 2.1.1-2. The evaluation here of these three combinations verifies that the fuel qualification flowchart provides a basis for selecting fuel that is appropriate not only for thermal and shielding limits, but also for confinement limits.

Fuel Description	Burnup	Enrichment	Cooling Time
Design Basis Fuel **(DBF-68)**	48 GWd/MTU	2.6 wt % U235	7 Years
Medium Burnup Fuel **(MBF-68)**	55 GWd/MTU	2.5 wt % U235	10 Years
High Burnup Fuel **(HBF-68)**	60 GWd/MTU	3.2 wt % U235	12 Years

The TN-68 cask is also authorized to load up to eight damaged assemblies with bundle average burnup ≤ 45 GWd/MTU at the peripheral locations of the cask. Fuel damage is limited in accordance with SAR Section 2.1, Table 8.1-1 step A14, and Technical Specification 2.1.1. The source terms for these damaged fuel assemblies are identical to those of the intact fuel assemblies.

Table 5.2-10 lists the activity representing the fission gases, volatiles, and fines contributing more than 0.1% of the activity contained in the 68 fuel assemblies, plus Iodine 129.

The releasable source term is first determined. The release fractions applied to the source term are provided below (developed from References 3 and 4).

Variable	Off-Normal Conditions	Accident Conditions
Fraction of crud that spalls off rods, f_C	0.15	1.0
Fraction of Rods that develop cladding breaches, f_B	0.10	1.0
Fraction of Gases that are released due to a cladding breach, f_G	0.3	0.3
Fraction of Fines that are released due to a cladding breach, f_F	3×10^{-5}	3×10^{-5}
Fraction of Released fines that remain airborne following a cladding breach, $F_{f,a}$ *	0.10	0.10
Fraction of Volatiles that are released due to a cladding breach, f_V	2×10^{-4}	2×10^{-4}

* 0.003% of the fuel in a rod is released from the rod during a cladding failure in the form of fines. However, only 10% of the fuel fines ejected from the rod during a cladding failure remain airborne (Reference 10).

The releasable source term also depends on the leak rate from the TN-68. Under off-normal conditions, it is assumed that the overpressure system is not functioning properly. In this case, the cask cavity gas is free to leak out at a rate of 1×10^{-5} std cc /sec. Assuming the cask cavity gas acts like helium (including the gases, volatiles, fines and crud), the leak rate is adjusted to a helium leak rate at cask cavity conditions using the equations of ANSI N14.5. This calculation is shown below.

P_u = 2.47 atm abs, 36.3 psig (off-normal cask cavity pressure assuming 10% of the fuel rods have failed–Section 4.7.5)
P_d = 1.0 atm abs
D = 4.825×10^{-4} cm
a = 0.5 cm
μ = 0.0279 cP (for helium at 479 K)
T = fluid absolute temp = average cavity gas temp = 402 °F = 479 K
M = 4.0
$P_a = \frac{1}{2}(P_u + P_d)$ = 1.735 atm abs

Substituting:

F_c = 9.674E-06
F_m = 5.399E-06
$L_{u,he}$ = 1.556E-05 cc/sec (conservatively 1.66E-05) of Helium for off-normal conditions

Similarly, under hypothetical accident conditions, it is assumed that the overpressure system has stopped functioning and fire conditions exist.

P_u = 5.89 atm abs, 71.7 psig (cask cavity pressure following hypothetical fire and assuming 100% fuel rod failure –Section 4.7.5)
P_d = 1.0 atm abs
D = 4.825×10^{-4} cm a = 0.5 cm
μ = 0.0296 cP (for helium at 573°K)
T = fluid absolute temp = average cavity gas temp following fire = 572 °F = 593K
M = 4.0
$P_a = \frac{1}{2}(P_u + P_d)$ = 3.44 atm abs

Substituting in to the equations of ANSI N14.5:

F_c = 9.119E-06
F_m = 2.978E-06
$L_{u,he}$ = 3.454E-05 cc/sec (conservatively 3.54E-05) of Helium for hypothetical accident conditions.

The releasable contents from the TN-68 during off-normal and hypothetical accident conditions are provided in Tables 7.3-1 and 7.3-2, respectively.

7.3.2 Release of Contents

Two scenarios are considered:

- Off-Normal Conditions – This condition exists over a one year period, seals are leaking at the test leak rate of 1 x 10^{-5} ref cm^3/s and the fraction of rods that have failed is 10%. Stability category D and 5 m/s wind speed is used for this analysis. This scenario assumes one cask is in off-normal condition at the ISFSI.

- Hypothetical Accident Conditions – This condition exists over a 30 day period, seals are leaking at the test leak rate of 1 x 10^{-5} ref cm^3/sec, the fraction of rods that have failed is 100%, and the temperature inside the cask is comparable to the fire accident conditions. Stability category F and 1 m/s wind speed is used for this analysis. This scenario assumes one cask is in the hypothetical accident condition at the ISFSI.

In the first scenario, the release is assumed to occur for more than a 20 minute period. The methodology of Reg Guide 1.145(5) is applied. The atmospheric diffusion from a ground level point source at 100 meters is based on the following parameters.

Wind speed = 5 meter/second
σ_y = 8 meters from Ref 5, Figure 1
σ_z = 5 meters from Ref 5, Figure 2
M = 1.1, from Ref 5, Figure 3
$\Sigma_y = M\sigma_y$ = 8.8 meters
A = is cross sectional area of the TN-68 = $12.6m^2$

Using the methodology of Reg Guide 1.145, $\{\chi/Q\}_{100\ \text{meters}}$ during off-normal conditions is 1.45E-03 sec/m^3. Similarly, the atmospheric diffusion for 500 meters during off- normal conditions is calculated using the following parameters.

Wind speed = 5 meter/second
σ_y = 40 meters
σ_z = 20 meters
M = 1.1
$\Sigma_y = M\sigma_y$ = 44 meters

During off normal conditions $\{\chi/Q\}_{500\ \text{meters}}$ is 7.23E-05 sec/m^3.

In the second scenario the release is assumed to be a short term ground level release (occurring however over a 30 day period) assuming the methodology of Regulatory Guide 1.25(6). The atmospheric stability classification of F and a wind speed of 1 m/sec is used. The atmospheric diffusion from a ground level point source at 100 meters is calculated below.

Wind speed = 1 meter/second
σ_y = 4 meters (Ref 5, Figure 1)
σ_z = 2.3 meters (Ref 5, Figure 2)

Substituting into the equations of Reference 6:

$\chi/Q = 1 / 1\ (\pi \times 4 \times 2.3) = 3.46E\text{-}02$ sec/m^3 for hypothetical accident conditions

Similarly, the atmospheric diffusion for 500 meters is:

Wind speed = 1 meter/second
σ_y = 20.0 meters (from Reference 6, Figure 1)
σ_z = 8.4 meters (from Reference 6, Figure 2)
$\{\chi/Q\}_{500\ meters}$ = 1.90E-03 sec/m^3 for hypothetical accident conditions.

7.3.2.1 Dose Calculations

Dose components are calculated following the method of Regulatory Guide 1.109[7] and utilizing dose conversion factors from EPA Federal Guidance Reports Numbers 11 and 12[8,9]. (Note: Two sets of DCFs depending upon the chemical state of Sr-90 are reported in Federal Guidance Report Number 11. One set of DCF values is for Sr in the form of $SrTiO_3$ and the other set is for Sr in all other forms. The Sr-90 fission product should not form $SrTiO_3$ within the storage cask and therefore the DCF for this compound was not used.)

To determine the committed doses (from air inhalation), the following equation is used:

$$Dose_{inhalation} = R \times \chi/Q \times Q \times DCF_{inhalation} \times Time$$

Where:
R = Inhalation Rate = 8,000 m^3/year = 2.54E-04 m^3/sec
χ/Q = Short term average centerline value of atmospheric diffusion for a ground level release (sec/m^3)
Q = amount of material released (µCi/sec)
$DCF_{inhalation}$ = Exposure Dose Conversion Factor (mrem/µCi), from reference 8.
Time = Time of Exposure (Seconds)

To determine the deep doses (from air immersion), the following equation is used:

$$Dose_{air\ immersion} = \{\chi/Q \times Q \times DCF_{air\ immersion}\} \times Time$$

Where:

χ / Q = Short term average centerline value of atmospheric diffusion for a ground level release (sec/m^3)
Q = amount of material released (µCi/sec)
$DCF_{immersion}$ = Exposure Dose Conversion Factor (mrem/year per µCi/cm^3), from ref 9
Time = Time of Exposure (Seconds)

For off-normal conditions, the estimated annual airborne doses (internal and external) at 100 meters from a single TN-68 cask are provided in Table 7.3-3. Since the DBF-68 fuel provides for the maximum source term (Co60 dominates the releases) for off-normal conditions, only the DBF-68 fuel is evaluated for confinement calculations. The deep dose (external) and the committed dose (internal) on an organ basis and total effective dose for distances of 100 and 500 meters are summarized below:

	Dose at 100 meters (mrem/yr)	Dose at 500 meters (mrem/yr)
Gonad	5.96E-01	2.97E-02
Breast	1.21E+00	6.02E-02
Lung	**2.25E+01**	**1.12E+00**
Red Marrow	3.14E+00	1.56E-01
Bone Surface	1.78E+01	8.88E-01
Thyroid	1.07E+00	5.36E-02
Remainder	2.87E+00	1.43E-01
Effective	**5.39E+00**	**2.69E-01**
Skin	5.16E-02	2.57E-03

The values presented in bold print above demonstrate that the criteria of 72.104(a) are met under off-normal conditions.

For hypothetical accident conditions, the committed doses (internal) and the deep doses (external) at 100 meters from a single TN-68 cask for a 30 day exposure is provided in Table 7.3-4. The total effective dose equivalent at 100 m and at 500 m from the TN-68 cask due to the three sources DBF-68, MBF-68 and HBF-68 is summarized in the tables below.

Target	**Dose at 100 meters (mrem)**		
	DBF-68	**MBF-68**	**HBF-68**
Gonad	2.04E+01	1.97E+01	1.87E+01
Breast	3.44E+01	2.36E+01	1.86E+01
Lung	6.49E+02	4.56E+02	3.69E+02
Red Marrow	1.16E+02	1.20E+02	1.20E+02
Bone Surface	**7.33E+02**	**8.98E+02**	**9.27E+02**
Thyroid	3.07E+01	2.12E+01	1.68E+01
Remainder	8.88E+01	7.53E+01	6.64E+01
Effective (TEDE)	**1.75E+02**	**1.50E+02**	**1.39E+02**
Skin	1.67E+00	1.26E+00	1.12E+00
Lens Dose (Skin + TEDE)	**1.76E+02**	**1.51E+02**	**1.40E+02**

Target	Dose at 500 meters (mrem)		
	DBF-68	**MBF-68**	**HBF-68**
Gonad	1.12E+00	1.08E+00	1.03E+00
Breast	1.89E+00	1.30E+00	1.02E+00
Lung	3.56E+01	2.50E+01	2.03E+01
Red Marrow	6.37E+00	6.57E+00	6.61E+00
Bone Surface	**4.02E+01**	**4.93E+01**	**5.09E+01**
Thyroid	1.69E+00	1.17E+00	9.22E-01
Remainder	4.88E+00	4.13E+00	3.65E+00
Effective (TEDE)	**9.60E+00**	**8.24E+00**	**7.64E+00**
Skin	9.18E-02	6.94E-02	6.15E-02
Lens Dose (Skin + TEDE)	**9.69E+00**	**8.31E+00**	**7.70E+00**

The maximum 30-day TEDE value of 175 mrem is due to the DBF-68 fuel. The corresponding 10CFR 72.106 limit is 5 rem.

The maximum 30-day Lens Dose Equivalent value of 176 mrem is also due to the DBF-68 fuel. The corresponding 10CFR 72.106 limit is 15 rem.

The maximum 30-day dose to any organ / tissue is 927 mrem and it occurs at the bone surface due to the HBF-68 fuel. The corresponding 10CFR 72.106 limit is 50 rem.

Therefore all the criteria of 72.106 are met at 100m.

For the accident conditions, the DBF-68 fuel provides for the maximum annual dose except bone surface and red marrow. For these organs, the Pu-238 and Cm-244 are important isotopes and their concentrations are higher at higher burnup.

A summary of the doses at 100m and their corresponding regulatory limits is shown below

Off-Normal Conditions		
Organ	**10CFR72.104(a) Limit (mrem)**	**Dose (mrem)**
Whole Body (TEDE)	25	5.39
Thyroid	75	1.07
Other Critical Organ	25	22.5 (Lung)

Accident Conditions		
Organ	**10CFR72.106(b) Limit (mrem)**	**Dose (mrem)**
Whole Body (TEDE)	5000	175
Organ (TODE)	50000	927 (Bone Surface)
Lens of Eye (LDE)	15000	176
Skin (SDE)	50000	1.67

7.3.2.2 Pressurization of Confinement Vessel

The cask cavity pressure for normal, off-normal, and accident conditions is calculated in Section 4.7.

7.3.3 Latent Seal Failure

By design the overpressure monitoring system does not immediately alarm if there is a leak in a seal or the overpressure system. The time period from when a leak begins to occur and when the overpressure system alarm is activated is dependent on the size of the leak. Two conditions which could exist within the TN-68 confinement system are:

(1) The outer seal (or the overpressure system) is leaking to the atmosphere. In this case the inner seal is intact and there is no release of the contents of the cask cavity to the atmosphere.

(2) The inner seal is leaking (or the overpressure system is leaking into the cask cavity). In this case the outer seal is still intact and there is no release of the cask cavity contents to the atmosphere.

If a latent seal leak has occurred, the tables below provide some examples of the time to alarm based on assumed leakage rates (and based on the conditions presented in Section 7.1.5).

Case 1 - Leakage of Overpressure System to the Atmosphere

Leak Rate (ref cm^3/s)	Estimated Time to Alarm (from Start of Latent Seal Failure)	Estimated Time to Loss of OP System Pressure (from Start of Latent Seal Failure)
1×10^{-3}	15 days	31 days
1×10^{-4}	160 days	326 days
5×10^{-4} (see Figure 7.1-2)	1 year	2.5 years
1×10^{-5} (see Figure 7.1-1)	11 years	over 20 years

Case 2 – Leakage of Overpressure System to Cask Cavity

Leak Rate (ref cm^3/s)	Estimated Time to Alarm (from Start of Latent Seal Failure)	Estimated Time to Equalize OP System Pressure with Cask Cavity Pressure (from Start of Latent Seal Failure)
1×10^{-3}	16 days	21 days
1×10^{-4}	175 days	220 days
5×10^{-4} (see Figure 7.1-2)	1.5 years	10 years
1×10^{-5} (see Figure 7.1-1)	15 years	over 20 years

As shown in the tables above, the alarm is set such that for any credible leak, there is time to evaluate the leaking condition and correct the condition provided that the overpressure system remains pressurized. This period can be extended by repressurizing the overpressure tank.

Another condition which has been considered is that a latent seal failure has occurred and the overpressure system is removed due to an accident.

(1) If the outer seal has the latent failure and the OP system is removed then there is no release of cask cavity contents to the atmosphere.

(2) If the inner seal has a latent failure and the OP system is removed then the table below provides the time before 10 CFR 72.106(b) limits will be exceeded (based on accident conditions presented in Section 7.2).

Standard Leak Rate (ref cm^3/sec)	Time to exceed 10 CFR 72.106(b) Limits
1×10^{-3}	8.5 days
1×10^{-4}	85 days
5×10^{-5}	171 days
1×10^{5}	857 days

The times above demonstrate that a latent failure up to 100 times greater than the test value could occur and recovery is possible.

The time to reach the accident release rates is dependent on the size of the leak. Due to the reliability of the metallic o-rings used in static applications, it is not considered credible that the inner seals could leak at a rate significantly higher than the test leak rate. The probability that a gross leak of an inner seal in combination with a gross leak in an outer seal or the overpressure system, such that the overpressure system could not hold pressure, is not considered a credible event.

However, if the overpressure system is not functional, the overpressure system can be replaced with a blind flange. The replacement of the overpressure system with the blind flange is described under contingency actions in Chapter 8, Section 8.4. The estimated operational dose due to this operation is provided in Chapter 10.

7.4 References

1. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Division 1 - Subsection NB, 1995 with Addenda through 1996.

2. ANSI N14.5-1997, “Leakage Tests on Packages for Shipment,” February 1998.

3. USNRC, Spent Fuel Project Office, Interim Staff Guidance No. 5, Revision 1.

4. NUREG-1617, “Standard Review Plan for Transportation Packages for Spent Nuclear Fuel, Draft Report for Comment” US Nuclear Regulatory Commission, March 1998.

5. USNRC Regulatory Guide 1.145,"Atmospheric Dispersion Models for Potential Accident Consequence Assessment at Nuclear Power Plants," Rev 1, 1983.

6. USNRC Regulatory Guide 1.25, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling Storage Facility for Boiling and Pressurized Water Reactors.”

7. USNRC Regulatory Guide 1.109, "Calculation of Annual Doses to Men from Routing Releases of Reactor Effluent for the Purpose of Evaluating Compliance with 10CFR50, Appendix I," Rev 1, 1977.

8. USEPA Federal Guidance Report No 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," EPA-520/1-88-0202, September 1988.

9. USEPA Federal Guidance Report No 12, "External Exposure to Radionuclides in Air, Water, and Soil," EPA-402-R-93-081, September, 1983.

10. SAND90-2406, “A Method for Determining the Spent Fuel Contribution to Transport Cask Containment Requirements,” Sandia National Laboratories, November 1992.

11. Helicoflex Catalog ET 507 E5930

TABLE 7.3-1

TN-68 Releasable Source Term for Off-Normal Conditions – Design Basis 8x8 Fuel (DBF-68)

Isotope	Activity (Ci/assembly)	Release Fraction	Concentration in Void Space of TN-68[1] (Ci/cm^3)	Material Released[2] Q (μCi/sec)
H 3	8.89E+01	0.30	3.02E-05	5.02E-04
Co 60[(3)]	5.04E+01	1.50E-01	8.57E-05	1.42E-03
Pu238	9.72E+02	3.00E-06	3.30E-09	5.49E-08
Pu239	5.15E+01	3.00E-06	1.75E-10	2.91E-09
Pu240	1.24E+02	3.00E-06	4.22E-10	7.00E-09
Pu241	1.90E+04	3.00E-06	6.46E-08	1.07E-06
Am241	2.78E+02	3.00E-06	9.45E-10	1.57E-08
Am243	1.35E+01	3.00E-06	4.59E-11	7.62E-10
Cm243	7.08E+00	3.00E-06	2.41E-11	4.00E-10
Cm244	2.22E+03	3.00E-06	7.55E-09	1.25E-07
Kr 85	1.28E+03	0.30	4.35E-04	7.22E-03
Sr 90	1.46E+04	2.00E-04	3.31E-06	5.49E-05
Y 90	1.46E+04	3.00E-06	4.96E-08	8.24E-07
Ru106	1.29E+03	2.00E-04	2.92E-07	4.85E-06
Rh106	1.29E+03	3.00E-06	4.39E-09	7.28E-08
Sb125	3.68E+02	3.00E-06	1.25E-09	2.08E-08
Te125m	8.98E+01	3.00E-06	3.05E-10	5.07E-09
I129	8.94E-03	0.30	3.04E-09	5.05E-08
Cs134	4.92E+03	2.00E-04	1.12E-06	1.85E-05
Cs137	2.47E+04	2.00E-04	5.60E-06	9.29E-05
Ba137m	2.33E+04	3.00E-06	7.92E-08	1.32E-06
Ce144	3.91E+02	3.00E-06	1.33E-09	2.21E-08
Pr144	3.91E+02	3.00E-06	1.33E-09	2.21E-08
Pm147	4.52E+03	3.00E-06	1.54E-08	2.55E-07
Eu154	1.04E+03	3.00E-06	3.54E-09	5.87E-08
Eu155	3.15E+02	3.00E-06	1.07E-09	1.78E-08

1 Values are based on 10% failure of the fuel rods and cask free volume of 6 m^3.

2 Values are based on 1.6E-05 cm^3 / sec helium leak from confinement.

3 The Co-60 source is calculated using the methodology of Reference 3. It is based on an 8x8 fuel assembly with surface area of 1601 cm^2/rod and a crud surface concentration of 1254 μCi / cm^2 at the time of discharge. (The value listed above includes a minimum cooling time of seven years.)

TABLE 7.3-2

TN-68 Releasable Source Term for Accident Conditions - Design Basis 8x8 Fuel (DBF-68)

Isotope	Activity (Ci/assembly)	Release Fraction	Concentration in Void Space of TN-68[1] (Ci/cm^3)	Material Released[2] Q (μCi/sec)
H3	8.89E+01	0.30	3.02E-04	1.07E-02
Co60[3]	5.04E+01	1.00E-00	5.71E-04	2.02E-02
Pu238	9.72E+02	3.00E-06	3.30E-08	1.17E-06
Pu239	5.15E+01	3.00E-06	1.75E-09	6.20E-08
Pu240	1.24E+02	3.00E-06	4.22E-09	1.49E-07
Pu241	1.90E+04	3.00E-06	6.46E-07	2.29E-05
Am241	2.78E+02	3.00E-06	9.45E-09	3.35E-07
Am243	1.35E+01	3.00E-06	4.59E-10	1.62E-08
Cm243	7.08E+00	3.00E-06	2.41E-10	8.52E-09
Cm244	2.22E+03	3.00E-06	7.55E-08	2.67E-06
Kr85	1.28E+03	0.30	4.35E-03	1.54E-01
Sr90	1.46E+04	2.00E-04	3.31E-05	1.17E-03
Y90	1.46E+04	3.00E-06	4.96E-07	1.76E-05
Ru106	1.29E+03	2.00E-04	2.92E-06	1.04E-04
Rh106	1.29E+03	3.00E-06	4.39E-08	1.55E-06
Sb125	3.68E+02	3.00E-06	1.25E-08	4.43E-07
Te125m	8.98E+01	3.00E-06	3.05E-09	1.08E-07
I129	8.94E-03	0.30	3.04E-08	1.08E-06
Cs134	4.92E+03	2.00E-04	1.12E-05	3.95E-04
Cs137	2.47E+04	2.00E-04	5.60E-05	1.98E-03
Ba137m	2.33E+04	3.00E-06	7.92E-07	2.80E-05
Ce144	3.91E+02	3.00E-06	1.33E-08	4.71E-07
Pr144	3.91E+02	3.00E-06	1.33E-08	4.71E-07
Pm147	4.52E+03	3.00E-06	1.54E-07	5.44E-06
Eu154	1.04E+03	3.00E-06	3.54E-08	1.25E-06
Eu155	3.15E+02	3.00E-06	1.07E-08	3.79E-07

1 Values are based on 100% failure of the fuel rods and cask free volume of 6 m3.

2 Values are based on 2.76E-05 cm3 / sec helium leak from confinement.

3 The Co-60 source is calculated using the methodology of Reference 3. It is based on an 8x8 fuel assembly with surface area of 1601 cm^2/rod and an initial crud surface concentration of 1254 $\mu Ci / cm^2$ at the time of discharge. (The value listed above includes a minimum cooling time of seven years).

TABLE 7.3-3

OFF-SITE AIRBORNE DOSES FROM OFF-NORMAL CONDITIONS AT 100M FROM THE TN-68 CASK

Design Basis 8x8 Fuel (DBF-68), Committed Doses (Internal) + Deep Dose (External) mrem/year

Isotope	Gonad	Breast	Lung	R. Marrow	B. Surface	Thyroid	Remainder	Effective
H3	3.73E-04	3.73E-04	3.73E-04	3.73E-04	3.73E-04	3.73E-04	3.73E-04	3.73E-04
Co60	3.21E-01	1.16E+00	2.11E+01	1.08E+00	8.68E-01	1.02E+00	2.23E+00	3.64E+00
Pu238	6.60E-02	2.36E-06	7.54E-01	3.58E-01	4.48E+00	2.27E-06	1.66E-01	2.50E-01
Pu239	3.97E-03	1.15E-07	4.03E-02	2.11E-02	2.64E-01	1.13E-07	9.44E-03	1.45E-02
Pu240	9.56E-03	2.86E-07	9.71E-02	5.08E-02	6.35E-01	2.72E-07	2.27E-02	3.49E-02
Pu241	3.14E-02	1.41E-06	1.47E-01	1.55E-01	1.94E+00	5.71E-07	6.04E-02	1.03E-01
Am241	2.19E-02	1.80E-06	1.24E-02	1.17E-01	1.46E+00	1.08E-06	5.27E-02	8.09E-02
Am243	1.07E-03	4.98E-07	5.83E-04	5.66E-03	7.11E-02	2.72E-07	2.53E-03	3.90E-03
Cm243	3.55E-04	1.08E-07	3.33E-04	2.03E-03	2.52E-02	6.62E-08	9.89E-04	1.43E-03
Cm244	8.56E-02	5.60E-06	1.04E-01	5.05E-01	6.30E+00	5.44E-06	2.57E-01	3.61E-01
Kr85	1.43E-04	1.64E-04	1.39E-04	1.33E-04	2.69E-04	1.44E-04	1.33E-04	1.46E-04
Sr90	6.23E-03	6.23E-03	8.81E-03	7.93E-01	1.72E+00	6.23E-03	1.35E-02	8.29E-01
Y90	3.63E-07	3.68E-07	3.30E-04	9.90E-06	9.91E-06	3.63E-07	1.37E-04	8.08E-05
Ru106	2.88E-03	2.86E-03	2.17E-01	2.86E-03	2.86E-03	2.86E-03	3.53E-03	2.69E-02
Rh106	1.25E-07	1.43E-07	1.25E-07	1.20E-07	2.12E-07	1.27E-07	1.19E-07	1.28E-07
Sb125	3.91E-07	4.51E-07	1.94E-05	6.45E-07	2.56E-06	3.60E-07	1.36E-06	3.02E-06
Te125m	2.75E-08	2.40E-08	2.27E-06	6.56E-07	6.99E-06	2.20E-08	1.47E-07	4.29E-07
I129	1.93E-07	4.59E-07	6.83E-07	3.05E-07	3.09E-07	3.38E-03	2.58E-07	1.02E-04
Cs134	1.06E-02	8.86E-03	9.62E-03	9.61E-03	9.13E-03	9.07E-03	1.13E-02	1.02E-02
Cs137	3.50E-02	3.13E-02	3.52E-02	3.31E-02	3.17E-02	3.17E-02	3.64E-02	3.45E-02
Ba137m	6.28E-06	7.17E-06	6.24E-06	6.08E-06	1.03E-05	6.41E-06	5.97E-06	6.41E-06
Ce144	1.83E-06	1.87E-06	7.50E-04	2.53E-05	4.31E-05	1.79E-06	9.77E-05	9.58E-05
Pr144	7.10E-09	8.05E-09	9.62E-08	7.07E-09	1.13E-08	7.30E-09	8.20E-09	1.84E-08
Pm147	2.38E-10	4.36E-10	8.49E-04	8.95E-05	1.12E-03	2.46E-10	6.46E-05	1.16E-04
Eu154	3.01E-05	3.98E-05	2.00E-04	2.68E-04	1.32E-03	1.86E-05	2.86E-04	1.96E-04
Eu155	2.79E-07	4.78E-07	9.10E-06	1.09E-05	1.16E-04	1.91E-07	8.49E-06	8.56E-06
Total	5.96E-01	1.21E+00	2.25E+01	3.14E+00	1.78E+01	1.07E+00	2.87E+00	5.39E+00

TABLE 7.3-3

OFF-SITE AIRBORNE DOSES FROM OFF-NORMAL CONDITIONS AT 100M FROM THE TN-68 CASK

(Continued)
Design Basis 8x8 Fuel (DBF-68), Deep Doses (External)
mrem/year

Isotope	Gonad	Breast	Lung	R. Marrow	B. Surface	Thyroid	Remain-der	Effective	Skin
H3	0.00E+00	0.00E+00	2.34E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.81E-08	0.00E+00
Co60	2.96E-02	3.35E-02	2.99E-02	2.96E-02	4.29E-02	3.06E-02	2.89E-02	3.04E-02	3.49E-02
Pu238	6.09E-11	1.18E-10	9.85E-12	1.56E-11	8.64E-11	3.73E-11	1.85E-11	4.53E-11	3.80E-10
Pu239	2.38E-12	3.72E-12	1.30E-12	1.31E-12	4.66E-12	1.91E-12	1.41E-12	2.09E-12	9.16E-12
Pu240	7.54E-12	1.46E-11	1.29E-12	1.96E-12	1.10E-11	4.65E-12	2.32E-12	5.63E-12	4.65E-11
Pu241	1.31E-11	1.57E-11	1.18E-11	1.02E-11	3.98E-11	1.27E-11	1.11E-11	1.32E-11	2.12E-11
Am241	2.28E-09	2.84E-09	1.79E-08	1.38E-09	7.63E-09	2.08E-09	1.68E-09	2.17E-09	3.40E-09
Am243	2.83E-10	3.37E-10	2.48E-10	2.00E-10	9.64E-10	2.70E-10	2.31E-10	2.81E-10	3.55E-10
Cm243	3.90E-10	4.52E-10	3.72E-10	3.38E-10	1.02E-09	3.90E-10	3.51E-10	3.98E-10	6.63E-10
Cm244	1.46E-10	2.82E-10	1.50E-11	3.10E-11	1.87E-10	8.89E-11	3.84E-11	1.04E-10	8.30E-10
Kr85	1.43E-04	1.64E-04	1.39E-04	1.33E-04	2.69E-04	1.44E-04	1.33E-04	1.46E-04	1.62E-02
Sr90	7.24E-08	8.83E-08	5.99E-08	5.06E-08	2.12E-07	6.82E-08	5.68E-08	7.01E-08	8.56E-05
Y90	2.64E-08	3.07E-08	2.47E-08	2.26E-08	6.20E-08	2.61E-08	2.34E-08	2.65E-08	8.71E-06
Ru106	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rh106	1.25E-07	1.43E-07	1.25E-07	1.20E-07	2.12E-07	1.27E-07	1.19E-07	1.28E-07	1.34E-06
Sb125	6.96E-08	7.98E-08	6.86E-08	6.58E-08	1.24E-07	7.07E-08	6.54E-08	7.11E-08	9.32E-08
Te125m	5.12E-10	7.28E-10	1.91E-10	1.60E-10	1.05E-09	3.98E-10	2.22E-10	3.89E-10	1.67E-09
I129	4.13E-09	5.69E-09	1.83E-09	1.40E-09	9.40E-09	3.30E-09	1.97E-09	3.25E-09	9.40E-09
Cs134	2.32E-04	2.64E-04	2.31E-04	2.25E-04	3.76E-04	2.37E-04	2.21E-04	2.37E-04	2.96E-04
Cs137	1.25E-07	1.52E-07	1.05E-07	8.97E-08	3.60E-07	1.19E-07	9.98E-08	1.22E-07	1.36E-04
Ba137m	6.28E-06	7.17E-06	6.24E-06	6.08E-06	1.03E-05	6.41E-06	5.97E-06	6.41E-06	8.31E-06
Ce144	3.19E-09	3.77E-09	2.87E-09	2.50E-09	9.31E-09	3.11E-09	2.70E-09	3.19E-09	1.10E-08
Pr144	7.10E-09	8.04E-09	7.10E-09	6.99E-09	1.12E-08	7.29E-09	6.88E-09	7.29E-09	3.15E-07
Pm147	3.23E-11	4.13E-11	2.35E-11	1.93E-11	9.42E-11	2.92E-11	2.27E-11	2.99E-11	3.50E-08
Eu154	5.96E-07	6.77E-07	5.95E-07	5.84E-07	9.37E-07	6.11E-07	5.72E-07	6.10E-07	8.24E-07
Eu155	7.50E-09	8.88E-09	6.68E-09	5.57E-09	2.44E-08	7.26E-09	6.23E-09	7.50E-09	1.02E-08
Total	3.00E-02	3.39E-02	3.02E-02	3.00E-02	4.35E-02	3.10E-02	2.93E-02	3.07E-02	5.16E-02

TABLE 7.3-4

OFF-SITE AIRBORNE DOSES FROM ACCIDENT CONDITIONS
AT 100M FROM THE TN-68 CASK

Design Basis 8x8 Fuel (DBF-68), mrem/30 Days, Committed Doses (Internal)

Isotope	Gonad	Breast	Lung	R. Marrow	B. Surface	Thyroid	Remainder	Effective
H3	1.56E-02	1.56E-02	1.56E-02	1.56E-02	1.56E-02	1.56E-02	1.56E-02	1.56E-02
Co60	8.11E+00	3.14E+01	5.88E+02	2.93E+01	2.30E+01	2.76E+01	6.14E+01	1.01E+02
Pu238	2.76E+00	9.86E-05	3.16E+01	1.50E+01	1.87E+02	9.49E-05	6.92E+00	1.05E+01
Pu239	1.66E-01	4.82E-06	1.69E+00	8.83E-01	1.10E+01	4.72E-06	3.95E-01	6.06E-01
Pu240	4.00E-01	1.20E-05	4.06E+00	2.13E+00	2.65E+01	1.14E-05	9.51E-01	1.46E+00
Pu241	1.31E+00	5.90E-05	6.13E+00	6.48E+00	8.10E+01	2.39E-05	2.52E+00	4.30E+00
Am241	9.17E-01	7.53E-05	5.19E-01	4.91E+00	6.12E+01	4.51E-05	2.21E+00	3.38E+00
Am243	4.46E-02	2.08E-05	2.44E-02	2.37E-01	2.97E+00	1.14E-05	1.06E-01	1.63E-01
Cm243	1.49E-02	4.52E-06	1.39E-02	8.48E-02	1.06E+00	2.75E-06	4.14E-02	5.96E-02
Cm244	3.58E+00	2.34E-04	4.35E+00	2.11E+01	2.63E+02	2.27E-04	1.08E+01	1.51E+01
Kr85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr90	2.61E-01	2.61E-01	3.68E-01	3.32E+01	7.18E+01	2.61E-01	5.66E-01	3.47E+01
Y90	1.41E-05	1.41E-05	1.38E-02	4.13E-04	4.12E-04	1.41E-05	5.73E-03	3.38E-03
Ru106	1.20E-01	1.20E-01	9.07E+00	1.20E-01	1.20E-01	1.20E-01	1.47E-01	1.13E+00
Rh106	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb125	1.34E-05	1.55E-05	8.10E-04	2.42E-05	1.02E-04	1.21E-05	5.41E-05	1.23E-04
Te125m	1.13E-06	9.75E-07	9.47E-05	2.74E-05	2.92E-04	9.05E-07	6.15E-06	1.79E-05
I129	7.88E-06	1.90E-05	2.85E-05	1.27E-05	1.25E-05	1.41E-01	1.07E-05	4.25E-03
Cs134	4.33E-01	3.59E-01	3.93E-01	3.93E-01	3.66E-01	3.69E-01	4.63E-01	4.16E-01
Cs137	1.46E+00	1.31E+00	1.47E+00	1.39E+00	1.33E+00	1.32E+00	1.52E+00	1.44E+00
Ba137m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce144	7.66E-05	7.81E-05	3.14E-02	1.06E-03	1.80E-03	7.46E-05	4.09E-03	4.01E-03
Pr144	9.56E-11	4.16E-10	3.73E-06	3.20E-09	5.35E-09	3.36E-10	5.55E-08	4.64E-07
Pm147	8.62E-09	1.65E-08	3.55E-02	3.74E-03	4.68E-02	9.08E-09	2.70E-03	4.86E-03
Eu154	1.23E-03	1.64E-03	8.36E-03	1.12E-02	5.52E-02	7.53E-04	1.19E-02	8.16E-03
Eu155	1.14E-05	1.96E-05	3.80E-04	4.57E-04	4.86E-03	7.67E-06	3.55E-04	3.58E-04
Total	1.96E+01	3.34E+01	6.48E+02	1.15E+02	7.31E+02	2.98E+01	8.80E+01	1.74E+02

TABLE 7.3-4

OFF-SITE AIRBORNE DOSES FROM ACCIDENT CONDITIONS
AT 100M FROM THE TN-68 CASK

(Continued)
HBF-68 Fuel, mrem/30 Days, Committed Doses (Internal)

Isotope	Gonad	Breast	Lung	R. Marrow	B. Surface	Thyroid	Remainder	Effective
H3	1.45E-02	1.45E-02	1.45E-02	1.45E-02	1.45E-02	1.45E-02	1.45E-02	1.45E-02
Co60	4.20E+00	1.62E+01	3.04E+02	1.52E+01	1.19E+01	1.43E+01	3.18E+01	5.22E+01
Pu238	3.78E+00	1.35E-04	4.32E+01	2.05E+01	2.56E+02	1.30E-04	9.47E+00	1.43E+01
Pu239	1.71E-01	4.96E-06	1.74E+00	9.09E-01	1.13E+01	4.86E-06	4.06E-01	6.24E-01
Pu240	4.29E-01	1.28E-05	4.36E+00	2.28E+00	2.85E+01	1.22E-05	1.02E+00	1.57E+00
Pu241	1.11E+00	4.97E-05	5.16E+00	5.45E+00	6.82E+01	2.01E-05	2.13E+00	3.62E+00
Am241	1.45E+00	1.19E-04	8.19E-01	7.75E+00	9.66E+01	7.13E-05	3.48E+00	5.34E+00
Am243	6.15E-02	2.87E-05	3.36E-02	3.26E-01	4.09E+00	1.56E-05	1.46E-01	2.25E-01
Cm243	1.71E-02	5.21E-06	1.61E-02	9.77E-02	1.22E+00	3.17E-06	4.77E-02	6.87E-02
Cm244	5.00E+00	3.27E-04	6.07E+00	2.95E+01	3.68E+02	3.18E-04	1.50E+01	2.11E+01
Kr85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr90	2.86E-01	2.86E-01	4.04E-01	3.64E+01	7.87E+01	2.86E-01	6.20E-01	3.80E+01
Y90	1.55E-05	1.55E-05	1.51E-02	4.53E-04	4.51E-04	1.55E-05	6.28E-03	3.70E-03
Ru106	4.83E-03	4.79E-03	3.64E-01	4.79E-03	4.79E-03	4.79E-03	5.91E-03	4.51E-02
Rh106	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb125	4.57E-06	5.28E-06	2.75E-04	8.23E-06	3.46E-05	4.11E-06	1.84E-05	4.18E-05
Te125m	3.82E-07	3.30E-07	3.21E-05	9.28E-06	9.90E-05	3.06E-07	2.08E-06	6.08E-06
I129	9.61E-06	2.31E-05	3.47E-05	1.55E-05	1.53E-05	1.72E-01	1.30E-05	5.19E-03
Cs134	1.11E-01	9.20E-02	1.01E-01	1.01E-01	9.37E-02	9.46E-02	1.18E-01	1.07E-01
Cs137	1.62E+00	1.45E+00	1.63E+00	1.53E+00	1.47E+00	1.46E+00	1.68E+00	1.59E+00
Ba137m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce144	1.05E-06	1.07E-06	4.29E-04	1.45E-05	2.46E-05	1.02E-06	5.59E-05	5.48E-05
Pr144	1.31E-12	5.70E-12	5.10E-08	4.39E-11	7.33E-11	4.60E-12	7.60E-10	6.35E-09
Pm147	2.42E-09	4.64E-09	9.97E-03	1.05E-03	1.31E-02	2.55E-09	7.59E-04	1.37E-03
Eu154	1.06E-03	1.40E-03	7.16E-03	9.58E-03	4.73E-02	6.45E-04	1.02E-02	6.99E-03
Eu155	6.75E-06	1.16E-05	2.26E-04	2.71E-04	2.88E-03	4.55E-06	2.11E-04	2.12E-04
Total	1.82E+01	1.81E+01	3.68E+02	1.20E+02	9.26E+02	1.63E+01	6.60E+01	1.39E+02

Table 7.3-4
Off-Site Airborne Doses From Accident Conditions at 100 M
From the TN-68 Cask

(Continued)
Design Basis 8x8 Fuel (DBF-68), mrem/30 Days, Deep Doses (External)

	Gonad	Breast	Lung	R. Marrow	B. Surface	Thyroid	Remain-der	Effective	Skin
H3	0.00E+00	0.00E+00	9.77E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.18E-06	0.00E+00
Co60	8.26E-01	9.34E-01	8.33E-01	8.26E-01	1.20E+00	8.53E-01	8.06E-01	8.46E-01	9.74E-01
Pu238	2.55E-09	4.94E-09	4.12E-10	6.53E-10	3.61E-09	1.56E-09	7.73E-10	1.90E-09	1.59E-08
Pu239	9.97E-11	1.55E-10	5.46E-11	5.50E-11	1.95E-10	7.99E-11	5.89E-11	8.73E-11	3.83E-10
Pu240	3.15E-10	6.10E-10	5.40E-11	8.18E-11	4.59E-10	1.94E-10	9.72E-11	2.35E-10	1.94E-09
Pu241	5.46E-10	6.59E-10	4.92E-10	4.28E-10	1.66E-09	5.30E-10	4.63E-10	5.51E-10	8.89E-10
Am241	9.54E-08	1.19E-07	7.49E-07	5.79E-08	3.19E-07	8.70E-08	7.05E-08	9.09E-08	1.42E-07
Am243	1.18E-08	1.41E-08	1.04E-08	8.37E-09	4.03E-08	1.13E-08	9.66E-09	1.18E-08	1.48E-08
Cm243	1.63E-08	1.89E-08	1.56E-08	1.42E-08	4.25E-08	1.63E-08	1.47E-08	1.66E-08	2.77E-08
Cm244	6.12E-09	1.18E-08	6.28E-10	1.30E-09	7.83E-09	3.72E-09	1.61E-09	4.36E-09	3.47E-08
Kr85	5.99E-03	6.86E-03	5.83E-03	5.58E-03	1.13E-02	6.04E-03	5.58E-03	6.09E-03	6.75E-01
Sr90	3.03E-06	3.69E-06	2.51E-06	2.12E-06	8.87E-06	2.85E-06	2.38E-06	2.93E-06	3.58E-03
Y90	1.10E-06	1.28E-06	1.03E-06	9.46E-07	2.59E-06	1.09E-06	9.81E-07	1.11E-06	3.64E-04
Ru106	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rh106	5.21E-06	5.98E-06	5.21E-06	5.03E-06	8.87E-06	5.31E-06	4.97E-06	5.36E-06	5.62E-05
Sb125	2.91E-06	3.34E-06	2.87E-06	2.75E-06	5.19E-06	2.96E-06	2.74E-06	2.97E-06	3.90E-06
Te125m	2.14E-08	3.04E-08	8.01E-09	6.68E-09	4.38E-08	1.67E-08	9.30E-09	1.63E-08	6.96E-08
I129	1.73E-07	2.38E-07	7.65E-08	5.86E-08	3.93E-07	1.38E-07	8.22E-08	1.36E-07	3.93E-07
Cs134	9.70E-03	1.11E-02	9.66E-03	9.43E-03	1.57E-02	9.93E-03	9.26E-03	9.93E-03	1.24E-02
Cs137	5.24E-06	6.37E-06	4.40E-06	3.75E-06	1.51E-05	4.97E-06	4.17E-06	5.10E-06	5.68E-03
Ba137m	2.63E-04	3.00E-04	2.61E-04	2.54E-04	4.31E-04	2.68E-04	2.50E-04	2.68E-04	3.47E-04
Ce144	1.33E-07	1.58E-07	1.20E-07	1.04E-07	3.89E-07	1.30E-07	1.13E-07	1.33E-07	4.58E-07
Pr144	2.97E-07	3.36E-07	2.97E-07	2.92E-07	4.67E-07	3.05E-07	2.88E-07	3.05E-07	1.32E-05
Pm147	1.35E-09	1.73E-09	9.85E-10	8.06E-10	3.94E-09	1.22E-09	9.51E-10	1.25E-09	1.47E-06
Eu154	2.49E-05	2.83E-05	2.49E-05	2.44E-05	3.92E-05	2.56E-05	2.39E-05	2.55E-05	3.45E-05
Eu155	3.14E-07	3.72E-07	2.80E-07	2.33E-07	1.02E-06	3.03E-07	2.61E-07	3.14E-07	4.27E-07
Total	8.42E-01	9.52E-01	8.49E-01	8.41E-01	1.22E+00	8.69E-01	8.21E-01	8.63E-01	1.67E+00

Table 7.3-4
Off-Site Airborne Doses From Accident Conditions at 100 M
From the TN-68 Cask

(Continued)
HBF-68 Fuel, mrem/30 Days, Deep Doses (External)

	Gonad	Breast	Lung	R. Marrow	B. Surface	Thyroid	Remain-der	Effective	Skin
H3	0.00E+00	0.00E+00	9.11E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.10E-06	0.00E+00
Co60	4.28E-01	4.83E-01	4.31E-01	4.28E-01	6.19E-01	4.42E-01	4.17E-01	4.38E-01	5.04E-01
Pu238	3.49E-09	6.75E-09	5.64E-10	8.93E-10	4.94E-09	2.13E-09	1.06E-09	2.59E-09	2.17E-08
Pu239	1.03E-10	1.60E-10	5.62E-11	5.66E-11	2.01E-10	8.22E-11	6.06E-11	8.98E-11	3.94E-10
Pu240	3.38E-10	6.54E-10	5.80E-11	8.77E-11	4.92E-10	2.08E-10	1.04E-10	2.53E-10	2.08E-09
Pu241	4.60E-10	5.55E-10	4.14E-10	3.60E-10	1.40E-09	4.46E-10	3.90E-10	4.64E-10	7.48E-10
Am241	1.51E-07	1.88E-07	1.18E-06	9.14E-08	5.04E-07	1.37E-07	1.11E-07	1.44E-07	2.25E-07
Am243	1.63E-08	1.94E-08	1.43E-08	1.15E-08	5.55E-08	1.55E-08	1.33E-08	1.62E-08	2.04E-08
Cm243	1.88E-08	2.18E-08	1.79E-08	1.63E-08	4.89E-08	1.88E-08	1.69E-08	1.92E-08	3.19E-08
Cm244	8.55E-09	1.65E-08	8.77E-10	1.81E-09	1.09E-08	5.19E-09	2.24E-09	6.09E-09	4.85E-08
Kr85	5.33E-03	6.11E-03	5.20E-03	4.97E-03	1.00E-02	5.38E-03	4.97E-03	5.42E-03	6.02E-01
Sr90	3.32E-06	4.05E-06	2.75E-06	2.32E-06	9.72E-06	3.13E-06	2.61E-06	3.21E-06	3.92E-03
Y90	1.21E-06	1.41E-06	1.13E-06	1.04E-06	2.84E-06	1.20E-06	1.07E-06	1.22E-06	3.99E-04
Ru106	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rh106	2.09E-07	2.40E-07	2.09E-07	2.02E-07	3.56E-07	2.13E-07	1.99E-07	2.15E-07	2.25E-06
Sb125	9.89E-07	1.13E-06	9.74E-07	9.35E-07	1.76E-06	1.00E-06	9.30E-07	1.01E-06	1.32E-06
Te125m	7.24E-09	1.03E-08	2.71E-09	2.26E-09	1.48E-08	5.64E-09	3.15E-09	5.51E-09	2.36E-08
I129	2.10E-07	2.90E-07	9.33E-08	7.15E-08	4.79E-07	1.68E-07	1.00E-07	1.66E-07	4.79E-07
Cs134	2.49E-03	2.83E-03	2.48E-03	2.41E-03	4.03E-03	2.54E-03	2.37E-03	2.54E-03	3.17E-03
Cs137	5.79E-06	7.04E-06	4.86E-06	4.15E-06	1.67E-05	5.49E-06	4.61E-06	5.63E-06	6.28E-03
Ba137m	2.91E-04	3.32E-04	2.89E-04	2.82E-04	4.78E-04	2.97E-04	2.76E-04	2.97E-04	3.85E-04
Ce144	1.82E-09	2.16E-09	1.64E-09	1.43E-09	5.33E-09	1.78E-09	1.55E-09	1.82E-09	6.27E-09
Pr144	4.06E-09	4.60E-09	4.06E-09	4.00E-09	6.40E-09	4.17E-09	3.94E-09	4.17E-09	1.80E-07
Pm147	3.80E-10	4.85E-10	2.77E-10	2.26E-10	1.11E-09	3.43E-10	2.67E-10	3.52E-10	4.12E-07
Eu154	2.14E-05	2.43E-05	2.13E-05	2.09E-05	3.36E-05	2.19E-05	2.05E-05	2.19E-05	2.95E-05
Eu155	1.86E-07	2.21E-07	1.66E-07	1.38E-07	6.05E-07	1.80E-07	1.55E-07	1.86E-07	2.53E-07
Total	4.36E-01	4.93E-01	4.39E-01	4.36E-01	6.34E-01	4.50E-01	4.25E-01	4.47E-01	1.12E+00

FIGURE 7.1-1

OVERPRESSURE MONITORING SYSTEM PRESSURE DROP WITH TIME
(Assuming Acceptance Test Leak Rate of 1×10^{-5} ref cm^3 /s)

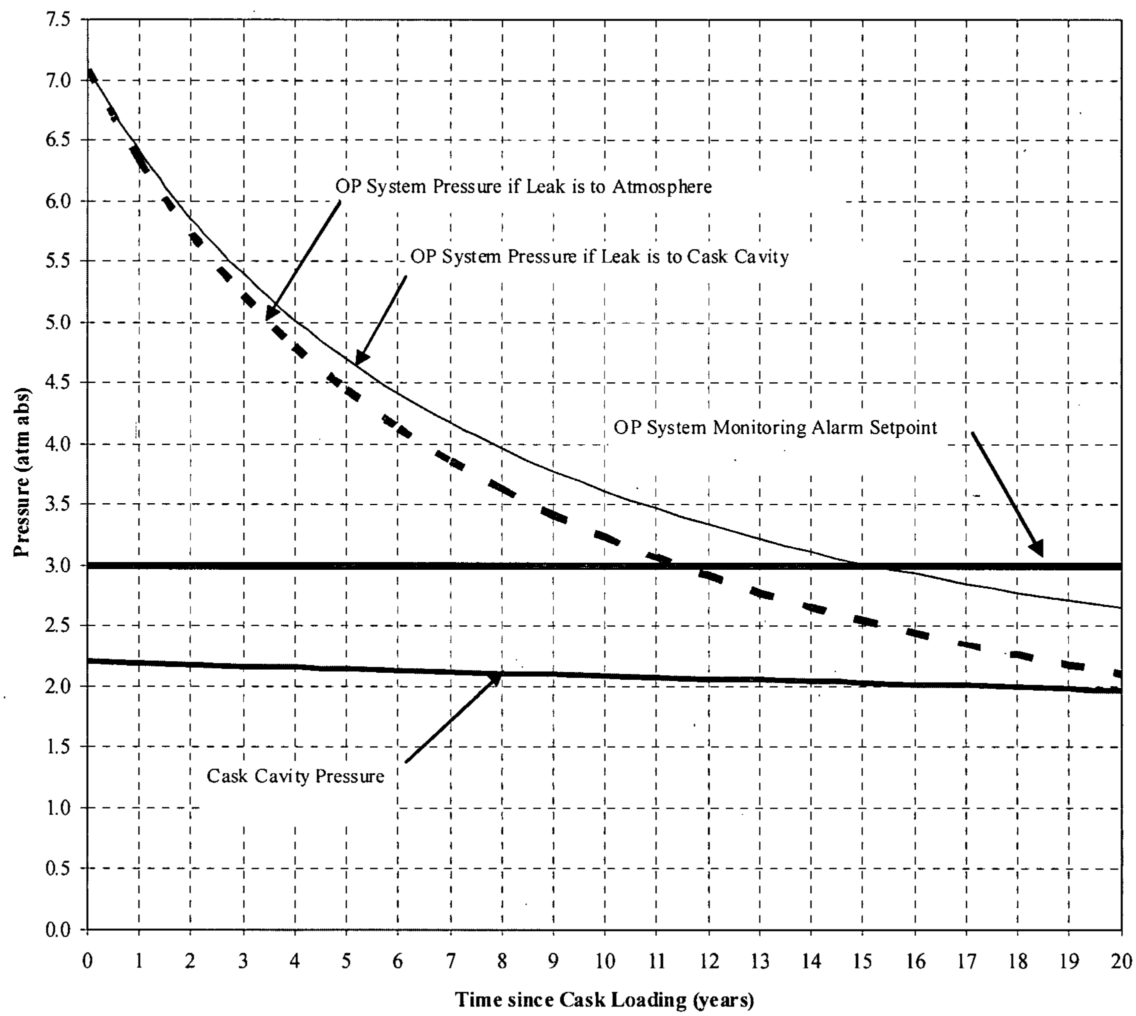


FIGURE 7.1-2

OVERPRESSURE MONITORING SYSTEM PRESSURE DROP WITH TIME
(Assuming a Latent Seal Leak Rate of 5 x 10^{-4} ref cm^3 /s)

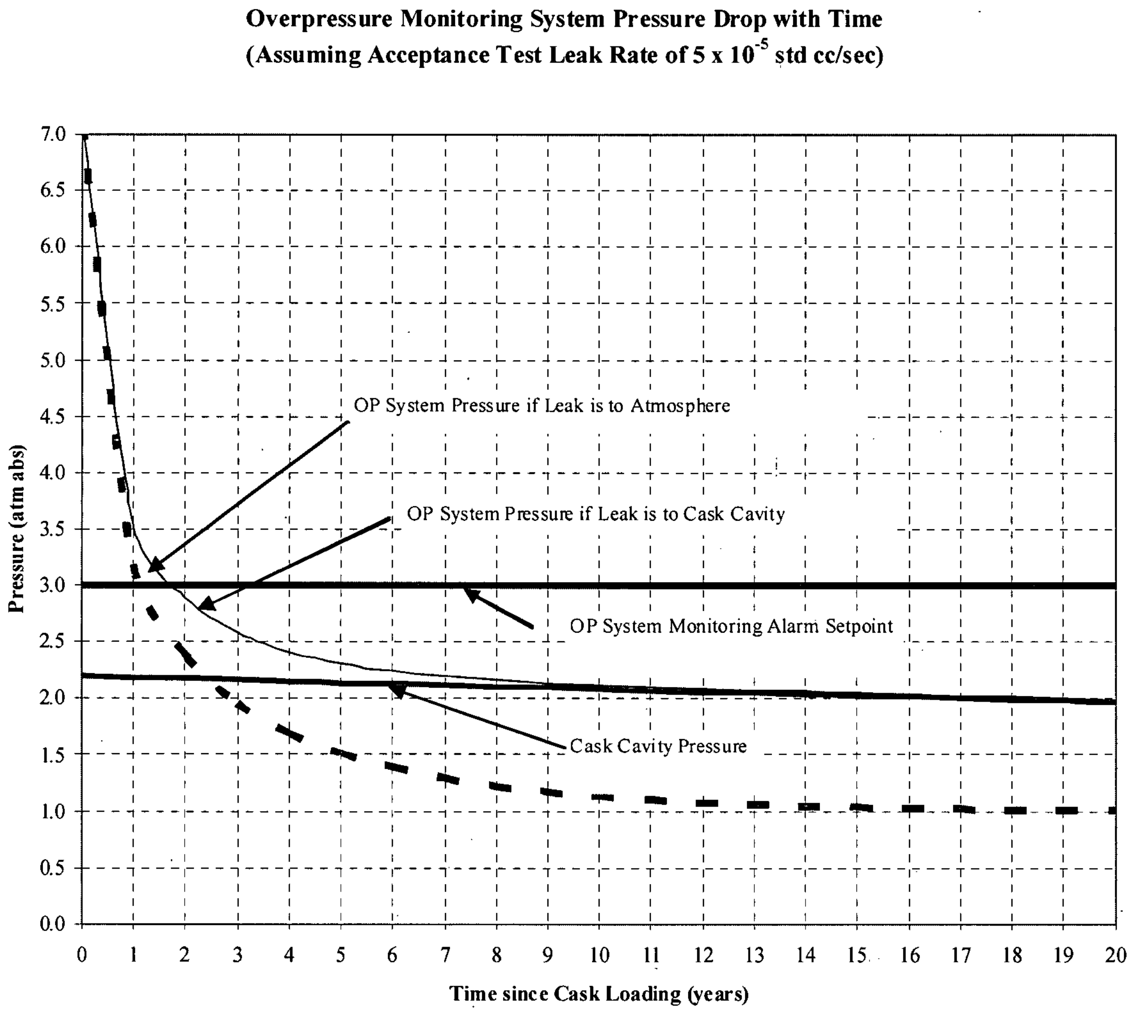
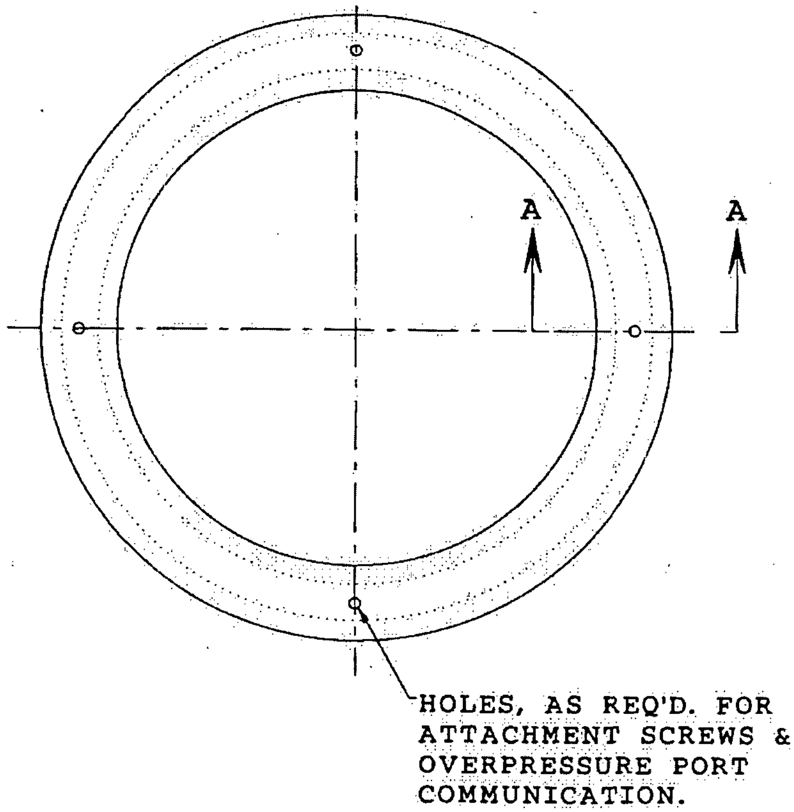
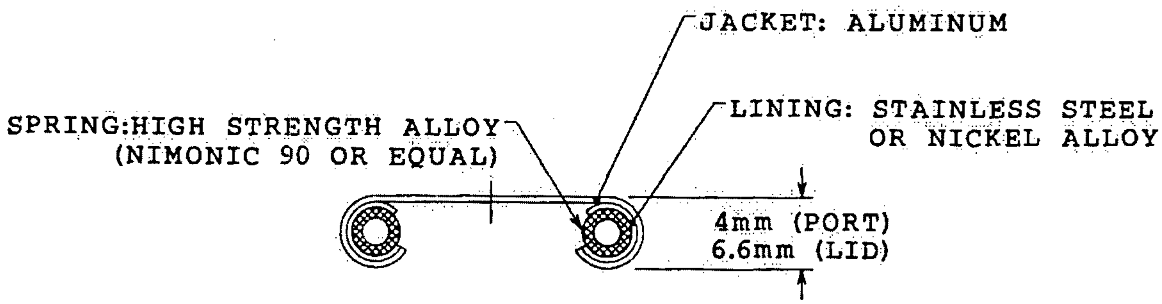


FIGURE 7.1-3
Lid, Vent Port and Drain Port Metal Seals





SECTION A-A

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CHAPTER 8

OPERATING PROCEDURES

This chapter outlines a sequence of operations to be incorporated into operating procedures for the preparation for and loading, testing, storing, unloading, and maintaining the TN-68 cask. Maintenance activities to be performed during the storage period are described in Chapter 9.

8.1 Loading the Cask

8.1.1 General Description

This section provides a general description of the cask loading operations. More detailed steps are provided in Table 8.1-1.

The empty casks will be receipt inspected. The protective cover, overpressure system, top neutron shield, and lid will be removed. The cask will be lowered into the spent fuel pool. As it is lowered, the cask will be filled with pool water or demineralized water. Fuel assemblies will be loaded into the cask using the refueling platform main hoist fuel grapple or equivalent methods (may vary depending on plant design).

After the cask is loaded with spent fuel and the lid is placed on the cask, the cask will be lifted to the pool surface, the water in the cask cavity will be drained and the lid bolts will be installed.

The cask will be set down and decontaminated. Because of its smooth surfaces the cask is designed to facilitate decontamination. The lid bolts will be torqued to their final value. The cask will be dried using a vacuum system. The cavity will be filled with helium to design pressure. and the cask lid seal will be leak tested. The top neutron shield will be installed on the lid. The overpressure monitoring system will be installed, and the interspaces between the double metallic o-rings pressurized. The external radiation levels will be measured. If the external radiation levels above the neutron shield exceed the values in Technical Specification 5.2.3, a shield ring shall be installed on the cask.

The protective cover will be installed and the cask will be transferred to its permanent storage location at the ISFSI. (The protective cover can be installed at the ISFSI.)

The cask will be transferred to the ISFSI site by a transport vehicle. The cask will be set in its storage position. The cask overpressure monitoring system will be connected and a functional check of the monitoring system will be performed.

8.1.2 Flow Sheets

The suggested sequence of operations to be performed in loading fuel into the TN-68 storage cask and placing the cask into storage at the ISFSI is outlined in Table 8.1-1. Some variations in this sequence may be expected after site specific procedures are developed by TN-68 users.

Details of the number of personnel and the time required for the various operations are given in Tables 10.3-1 and –2 as part of the radiation exposure determinations discussed in Chapter 10. The data is based on Transnuclear's experience with transport cask operations and will vary for an individual licensee. Temporary shielding, measures to facilitate surface decontamination and minimization of operation time will maintain operational doses ALARA as discussed in the flow sheets.

8.1.3 Vacuum Drying System

A vacuum drying system is utilized to remove residual moisture from the cask cavity, after the cask has been drained. This method is successfully used by Transnuclear on both its transport casks and storage casks.

After a loaded cask is removed from a pool and drained, it is placed under a vacuum. After bolting the lid, residual water is removed by the following or equivalent method:

a) Using a wand attached to the vacuum system, remove excess water from the seal areas through the passageways at the overpressure, drain and vent ports.
b) Remove the quick disconnect from the drain port, and install the drain port cover.
c) With the quick disconnect removed to improve evacuation, install a flanged vacuum connector over the vent port, purge or evacuate the helium supply lines, and evacuate the cask to 4 millibar (4×10^{-4} MPa) or less. Make provision to prevent or correct icing of the evacuation lines.
d) Isolate the cask by closing at least one valve between the cask and the vacuum pump, and either
   1. Shut off the vacuum pump,
   2. Disconnect the vacuum pump from the vacuum line to the cask, or
   3. Vent the vacuum line outside the first valve from the cask to atmosphere.

   If, in a period of 30 minutes, the pressure does not exceed 4 millibar (4×10^{-4} MPa), the cask is adequately dried. Otherwise, repeat vacuum pumping until this criterion is met.
e) Backfill the evacuated cask cavity with helium (minimum 99.99% purity) to slightly above atmospheric pressure, remove the vacuum connector, and immediately install the quick disconnect fitting.
f) Attach the vacuum/backfill manifold to the fitting, purge or evacuate the helium supply lines, and re-evacuate the cask to below 100 mbar.
g) Isolate the vacuum pump, backfill the cask cavity to above atmospheric pressure with helium (minimum 99.99% purity), and leak test. (See Section 8.1.4).

The evacuation and backfill process is repeated if the cask cavity is exposed to the atmosphere.

8.1.4 Leak Detection

After backfill, the cask is leak tested by helium mass spectrometry by pressurizing the annular space and measuring the total leak rate of all seals, both inner and outer, including the overpressure system. This conservative leak rate must be less than 1×10^{-5} ref-cm^3/sec (1.0×10^{-5} mbar-l/sec). Leak test procedures make provision for cases where a quick disconnect fitting may prevent communication between the cask cavity and the inside of a port inner seal.

Failure to meet the leak test acceptance criterion requires evaluation of the leak location, for example by the use of the helium mass spectrometer in the "sniffer" mode, examination of sealing surfaces, replacement of the leaking seal(s), and re-performance of the leak test. Replacement of the main lid seal requires reflooding of the cask and removal of the lid, similar to the steps described under Section 8.2.

8.1.5 Major Tools and Equipment

The following tools and equipment are normally required for loading and unloading the TN-68 casks:

- A transport frame which is used to transport the empty cask from the manufacturer's facility to the utility. The transport frame is not important to safety, since it is only used in conjunction with an empty cask.
- A spreader lift beam to connect the cask to the crane hook. The lift beam is used to remove the cask from the transport frame, to move the cask into the pool, into the processing stations such as the decontamination area and eventually to a location where the cask can be lifted by the cask transporter. This lift beam is designed and fabricated in accordance with ANSI N14.6.[(1)] The load bearing components of the lift beam are evaluated by the user under its heavy lifting program in accordance with NUREG-0612[(3)].
- A vertical cask transporter. The cask transporter is set to ensure that the loads from a postulated drop accident will be bounded by the maximum analyzed loads and given in Technical Specifications 4.1.2 and 5.2.2. The cask transporter is used to move the cask from the cask loading bay to the storage pad or from the pad back to the plant. The cask transporter may be self-propelled or be pulled by a tow vehicle to the ISFSI. The cask transporter is not important to safety, since the cask is analyzed to withstand an 18 inch drop onto a concrete pad which is bounding for the transfer path. The cask transporter is designed to lift the cask by means of the top trunnions.
- A lid lifting system. This may consist of a set of slings threaded into the top of the lid or a lifting pintle. The load bearing components of the lid lifting system are evaluated by the user under its heavy lifting program in accordance with NUREG-0612.
- Helium leak detector including port connectors. The leak detector is designated as not important to safety, but will be calibrated.
- Vacuum drying system including hoses and connectors. The vacuum drying system is designated as not important to safety, but all appropriate gages will be calibrated.
- Pumps for removing water from the cask. The pumps are not important to safety.
- Calibrated torque wrenches for setting specified torque for cask bolts, screws and plugs (Not important to safety).

- Sockets and hex keys for removal and replacement of bolts, screws, coupling and connectors. These items are not important to safety.
- Helium cylinders and manifold with calibrated pressure gage for backfill of cask and overpressure system. These items are not important to safety.
- Temporary blind flange which can be used to replace the overpressure port cover for transfer of the cask to the spent fuel pool.

8.2 Unloading the Cask

This section describes the steps required to unload a TN-68 cask. Additional measures may need to be taken if damage to the cask has occurred due to accidents.

If the TN-68 cask needs to be unloaded for any reason, the sequence of operations described in Section 8.1 and listed in Table 8.1-1 will be essentially performed in reverse. The unloading steps are provided in Table 8.2-1.

The dry cask reflood process during unloading of BWR fuel has the potential to disperse crud into the fuel pool and the pool area atmosphere, thereby creating airborne exposure and personnel contamination hazards. Radiation monitoring will be required during reflooding operations. Site specific procedures will be prepared prior to first use of the cask to address these concerns.

If the overpressure system is known to be leaking and no longer above cavity pressure, the cask overpressure monitoring system is disconnected.

The cask will be moved from the ISFSI back into the spent fuel pool building using the cask transporter. The protective cover will be unbolted and removed. The overpressure system will be depressurized. The overpressure port flange, the overpressure tank and top neutron shield will be removed. The vent port cover will be removed and a cavity gas sample will be collected. The gas sample will be analyzed and any precautions necessary will be added based on the cavity gas sample results.

If degraded fuel is suspected, additional measures, appropriate for the specific conditions, are to be planned, reviewed and approved by appropriate plant personnel, and implemented to minimize exposures to workers and radiological releases to the environment. These additional measures may include provision of filters, respiratory protection and other methods to control releases and exposures ALARA.

The helium in the cavity will be depressurized to atmospheric pressure. The drain port cover will be removed. The lid lifting equipment will be attached and the lid bolts untorqued. Remove some of the lid bolts, but keep at least 6 equally spaced lid bolts installed.

Fill and drain lines are connected to the lid drain and vent ports. The quick disconnect fittings may be used or they may be removed. The cask may be filled before lowering the cask into the pool or with the cask partially submerged in the spent fuel pool.

Pool water or demineralized water will be added to fill the cask and to gradually cool the fuel in the cask. (See Figure 8.2-1). The pressure is monitored at the cask outlet, and the flow rate of the water is controlled to limit the internal pressure to below the design limit of 100 psig (114.7 psia). A flow restriction valve will be installed at the inlet to the cask to restrict cooling water flow if the cask pressure exceeds the inlet water pressure (90 psia max.). The initial flow rate

will be set at about 3.0 gallon per minute. With the vent port quick connect fitting removed per Table 8.2-1 step 11, once the pressure falls below 60 psig and is maintained for a period of thirty five minutes, the flow rate can then be gradually increased while monitoring the pressure at the outlet. If the pressure gage reading exceeds 65.6 psig, close the inlet valve until the pressure falls below 60 psig. Reflooding can then be resumed.

The water/steam mixture from the vent port discharge may contain some radioactive material. Gases shall be closely monitored to determine if there is a radiological hazard and appropriately processed. A typical set up for filling the cask is shown in Figure 8.2-1. The flow restriction valve and the monitoring of the exit pressure will ensure that the water vapor pressure generated during unloading does not exceed the cask design pressure.

When the cask is full of water, the fill and drain lines will be removed. The remaining lid bolts will be removed. The cask will then be lowered to the pool bottom where the lid would be removed making the fuel accessible for transfer.

Provided that the TN-68 cask is within its design life, the cask can be reused after unloading. Inspection procedures should verify that the cask is still in its design configuration after unloading.

The TN-68 cask is designed so that it will not need to be opened after it has been closed and leak tested until it is time to unload the fuel.

8.3 Surveillance and Maintenance

Chapters 9 and 12 discuss required surveillance and maintenance of the TN-68 cask. Most required activities are very simple and do not require additional detail here. The most complex surveillance and maintenance operation is overpressure system maintenance, which is discussed below.

The term "switches" in the following refers to switches or transducers, either of which are used to monitor the pressure in the overpressure tank.

Redundant overpressure system switches are mounted on the side of the cask, and communicate with the overpressure tank via stainless steel tubing which penetrates the weather protective cover. Each switch has an isolation valve and an access valve provided for the calibration and maintenance procedure. The access valve outside port may have a capped fitting or a quick connect fitting.

To verify the functioning of the switches, a Channel Operational Test (COT) shall be performed. A helium pressure source and the appropriate test equipment is required. A typical procedure outline is provided below.

a) Close the isolation valve.

b) Remove the cap from the access valve, and connect the test equipment while maintaining a slow helium purge.

c) Pressurize test manifold to about 75 psig from the helium cylinder, then isolate the helium source and open the access valve.

d) Open the bleed down valve, and reduce the pressure slowly (Radioactive gases are not expected. However, provisions should be made to prevent any potential releases). For transducers, verify the pressure reading on the transducers against the reference gauge at a number of points. For both switches and transducers, verify that the alarm is actuated at the correct pressure.

e) Adjust the set point or calibrate as required and repeat the above test.

f) Repeat the procedure for the second switch if in service.

g) Repressurize the manifold with helium to the original system pressure (73.5 psig), close the access valve, disconnect the test equipment, cap the access valve, and open the isolation valve.

h) If replacement of a switch is required, the switch must be leak tested after installation.

If there has been some reduction in system pressure, the entire overpressure system may also be

re-pressurized to the original pressure during the channel operational test (73.5 psig) by opening the isolation valve at the beginning of step g) rather than at the end. The overpressure system may also be repressurized independently of a COT.

There is no requirement for periodic inspection or replacement of the elastomer o-ring seal on the protective cover. However, if any maintenance operation requires removal of the cover, the o-ring should be inspected at that time. If there are any signs of deterioration (hardening, cracking, permanent set) it should be replaced.

8.4 Contingency Actions

Routine surveillance activities may trigger contingency actions as identified in the Technical Specifications. Many of these actions, such as removal of storm debris, are simple and require no further detail here. This section provides guidance in the event of a low pressure alarm from the overpressure monitoring system. The margin between the set point and the confinement pressure provides ample time as provided in the Technical Specifications to assess and correct the condition.

First determine if there is a false indication. This could be due to alarm panel malfunction or a switch failure. Exceptionally cold weather may also cause a reduction in pressure and a consequent false alarm. This may be corrected by re-pressurizing the system as discussed at the end of Section 8.3.

If the alarm appears to be due to an actual leak, determine if there is a leak in the overpressure system. This may be done by checking the exterior plumbing, and then, if no leak is found, by removing the weather cover, and testing the tank and the overpressure port cover. A helium mass spectrometer system in either vacuum or sniffer mode may be used. If a leak is found, the overpressure system should be vented to atmospheric pressure to allow for repair work. Capture the helium in an evacuated cylinder to minimize the chance of radioactive effluents, and to provide a sample for testing. The overpressure system can then be repaired, reassembled, leak tested, and repressurized. A failure of the overpressure system for a period of 30 days has been evaluated as an off normal event. This should provide sufficient time to perform any repairs and testing. A temporary blind flange may be installed on the overpressure port during the repair.

If the alarm is not false, and there is no leak in the overpressure system, there may be a leak at the lid seal or the two port seals. Replacement of these seals will require returning the cask to an appropriate containment building.

After transfer, remove the weather cover, neutron shield, and the vent port. Vent the cavity to atmospheric pressure via the quick-connect coupling in the vent port. Capture a portion of the vented gas in a sample cylinder for analysis and vent the remainder to an appropriate area. Remove the drain port cover. Inspect the sealing surfaces and replace the seals and if necessary the covers. Repressurize the cask, assemble the port covers and leak test as required.

If after these steps, the cask still does not meet the leak tightness criterion, the lid gasket may be replaced. Proceed as for cask unloading, Section 8.2, up to the point of removing the lid. After the lid is removed and the fuel off-loaded, inspect the sealing surfaces, replace the seal, and reassemble the cask, proceeding in the normal sequence of loading operations.

8.5 Preparation of the Cask

The operations required for preparing the cask for transfer to the storage pad are provided in Table 8.1-1, Section C and D.

The following procedural steps shall be verified before moving the cask to the storage pad:

- The lid and penetration covers have been installed and torqued to their specified values.
- The cask has been vacuum dried and successfully dryness tested.
- The cask has been leak tested to ensure that the total leakage rate of both inner and outer seals is less than 1 x 10^{-5} ref-cm^3/sec (1.0 x 10^{-5} mbar-l/sec).
- The cask cavity has been backfilled to 2.0 atm abs (14.7 psig) with helium. The overpressure system has been backfilled to achieve an equilibrium pressure of about 6 atm abs (73.5 psig) with helium.
- The cask outside surfaces have been decontaminated. The surface contamination levels have been measured and do not exceed 20 dpm/ 100 cm^2 (alpha) or 1000 dpm/ 100 cm^2 (beta + gamma).
- The surface dose rates have been measured and do not exceed the technical specification limits provided in Chapter 12.

8.6 References

1. ANSI N14.6, "American National Standard for Radioactive Materials Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More," New York, 1986.

2. ANSI N14.5-1997 , "Leakage Tests on Packages for Shipment of Radioactive Materials," February 1998.

3. US Nuclear Regulatory Commission "Control of Heavy Loads at Power Plants, NUREG-0612, July, 1980.

TABLE 8.1-1
SEQUENCE OF OPERATIONS - LOADING

A. Receiving

1. Verify that records for the cask are complete and accurate and unload empty cask and seals at plant site.

2. Inspect for shipping damage. Check for shipment completeness and cleanliness.

3. Remove protective cover, overpressure system and top neutron shield.

4. Remove neutron shield pressure relief valve and install plug in neutron shield vent hole.

5. Remove lid bolts and lid.

Steps 6 through 11 may be performed in any order.

6. Replace lid seal by attaching new seal to lid by means of retaining screws. Inspect the lid sealing surface. Check for defects in the seal contact areas that may prevent a proper seal. (This step may be performed at any time prior to installing lid on loaded cask).

7. Replace seals in vent, drain and o.p. cover. Inspect the sealing surfaces. Check for defects in the seal contact areas that may prevent a proper seal. (This step may be performed at any time prior to installing covers on the loaded cask.)

8. Inspect cask for foreign material and handle, as appropriate.

9. Visually inspect the lid bolts and bolt hole threads to ensure they do not have any laps, seams, cracks or damaged threads.

10 Verify installation of a threaded plug in the vent hole of the neutron shield.

11. To minimize contaminants introduced into the spent fuel pool, with a clean hose, rinse the interior and exterior of the cask with demineralized water, if necessary.

12. Move cask to cask loading pool area.

13. At any time prior to loading, verify the basket type (B10 areal density in neutron absorber and outfitting for damaged fuel) from the cask serial number.

14. At any time prior to loading, verify that the fuel assemblies to be loaded meet the criteria of Technical Specification 2.1.1, corresponding to the basket type.

TABLE 8.1-1 (continued)
SEQUENCE OF OPERATIONS

Make an assessment of candidate fuel bundles regarding fuel integrity and handling. Review station operating, maintenance and/or corrective action records to ensure that candidate bundles conform to TN-68 Technical Specification 2.1.1 and have sufficient fuel cladding integrity such that fuel pellets are not expected to be released during normal handling.

If the records do not indicate leakage through the cladding, and no unusual conditions are evident, then the fuel may be classified as intact (undamaged).

If the records indicate leakage or other damage, then the fuel must be considered damaged unless additional analysis of the information demonstrates the nature of the damage is small. Such analyses could consider the in-core radionuclide differences between small, pinhole-like cladding leaks and larger cladding breaches.

Fuel lacking adequate records should be further examined to demonstrate its condition. Testing methods such as sipping, UT, etc., may be used to evaluate the fuel for cladding damage. If the examination results indicate that the fuel has cladding damage greater than pinhole leaks or hairline cracks, it is classified as damaged.

15. Damaged fuel may only be loaded in the eight outermost compartments of a basket outfitted with damaged fuel compartment extensions. If damaged fuel is to be loaded, verify that damaged fuel bottom end caps are installed in those compartments that will be loaded with damaged fuel. This may be done at any time prior to loading the fuel.

TABLE 8.1-1 (continued)
SEQUENCE OF OPERATIONS

B. Cask Loading Pool

1. Lower cask into cask loading pool while rinsing exterior of cask with demineralized water and fill interior with demineralized water or pool water.

2. Load preselected spent fuel assemblies into the basket compartments.

3. Verify identity of the fuel assemblies loaded into the cask. Install the top end caps on any compartments loaded with damaged fuel. Install hold down ring.

4. At least one lid penetration must be completely open (both cover and quick disconnect fitting removed) prior to installation of the lid. Lower lid and place on cask body flange over the two alignment pins.

5. Lift cask so that the top of the cask is above the water surface of pool and install at least six of the lid bolts. The lid bolts should be hand tight.

Note: Throughout this procedure, all bolt threads are to be coated with Nuclear Grade Neolube or equivalent.

6. Using the drain port in the lid, drain water from the cask in accordance with procedures. This may be done either before or after lifting the cask out of the pool. While lifting the cask out of the pool, the cask may be rinsed with clean deionized water to facilitate decontamination. If required to stay under crane load limits, water may be drained from the cask before lifting it clear of the water.

Note: Take measures as required to mitigate hydrogen accumulation in the cask as described in Section 3.4.1.4. For example, at least one lid port should remain open while there is water in the cask, unless the cask cavity is purged with helium or nitrogen.

One of the following four actions shall be taken to prevent oxidation of fuel pellets:

A. The cask may be loaded with only fuel that has no through-cladding defects, including pinholes or hairline cracks.

B. Draining of water may take place in a continuous operation followed shortly by vacuum drying. That is, cask operations subsequent to draining, e.g., completing the transfer of the cask from the pool and torquing the bolts, are to continue without delay, i.e., work around the clock, until vacuum drying is in progress.

C. Draining of water may be accompanied by a purge of helium or nitrogen to provide inert gas cover to the fuel, in which case the draining need not be a continuous operation as described in option B.

TABLE 8.1-1 (continued)
SEQUENCE OF OPERATIONS

D. A steady state or transient thermal analysis of the cask, considering the actual thermal load of the fuel and locations where any fuel with through-cladding defects is loaded, may be performed to demonstrate that the conversion of UO_2 to U_3O_8 will not be significant enough to cause further cladding damage from expansion of the fuel in rods with such defects.

7. Disconnect hose(s) from the port(s).

8. Move cask to the decontamination area.

TABLE 8.1-1 (continued)
SEQUENCE OF OPERATIONS

C. Decontamination Area

Note: The maximum potential for worker exposure occurs during decontamination and for operations near the lid from the time that the water in the cask is pumped out until the time the neutron shield is in place, steps C1 through C7. Exposure can be minimized by use of temporary shielding (lead "bean bags", plastic neutron shielding), by measures to facilitate decontamination, and by minimizing time and maximizing distance. A shield ring is provided to reduce the dose rates on the side of the cask above the neutron shield.

1. Initiate decontamination of the cask until acceptable surface contamination levels are obtained.

Note: Previously installed bolts may be removed for drying of bolt holes if at least six bolts remain installed at all times.

2. Install remaining lid bolts and torque lid bolts to the value specified on Drawing 972-70-1. This should be accomplished in multiple passes in accordance with an appropriate torquing pattern. Perform a final pass to ensure proper torque. A circular pattern may be utilized to eliminate further bolt movement.

   If the drain port quick connect fitting was removed for draining, it need not be reinstalled. If the fitting is installed, the space between the fitting and the cover must be purged with helium as described in step 6 below for the vent port. Install the drain port cover and tighten the bolts to the value specified on drawing 972-70-1.

3. Remove plug from neutron shield vent and reinstall pressure relief valve.

4. Connect the Vacuum Drying System (VDS) to the vent port and establish a vacuum to evaporate residual cavity water. Limit the rate of evacuation or provide a heat source such as heat tape on the evacuation line as necessary to prevent blockage of the line by ice.

5. Evacuate cavity to remove remaining moisture and verify dryness in accordance with Section 8.1.3.

6. Backfill cask with helium and pressurize to 2.0 atm abs (14.7 psig). Install the vent port cover, purging the cavity below the cover with helium at a minimum flow rate of 80 cubic feet per hour for at least 20 seconds. A partial pressure of at least 50% helium will be obtained under the cover. Tighten the vent port cover bolts to the value specified on drawing 972-70-1.

TABLE 8.1-1 (continued)
SEQUENCE OF OPERATIONS

7. Helium leak test all lid and port cover seals. The acceptable total cask seal leakage (both inner and outer seals) is 1 x 10^{-5} ref-cm^3/sec (1.0 x 10^{-5} mbar-l/sec). The leak test shall be performed in accordance with ANSI N14.5[(2)].

    7A. If cask does not pass leak test, determine source of leak. If the leak is in a vent or drain cover, remove the cover and replace the seals. Also examine the sealing surface for any obvious indication of scratches or defects. Repeat leak test.

    7B. If the cask still does not pass leak test, evaluate test method or return cask to pool and replace seals.

    7C. At the option of the user, leak testing may be deferred until assembly of the overpressure system is completed.

8. Install top neutron shield.

Note: Installation of the overpressure system and protective cover could be done at a different location if restricted overhead clearances require transfer without these components in place. A temporary blind flange and metal seal will be installed on the overpressure port prior to transferring the cask without the overpressure system in place. Temporary weather protection will be provided as necessary.

9. Install overpressure system tank and port flange. The o.p. cover bolts should be torqued to the value specified on drawing 972-70-1.

Note: At the option of the user, leak testing of the overpressure system and subsequent re-pressurization of the overpressure system (step 11) may be deferred until the entire system is assembled (see steps 14 and 15).

10. Helium leak test the overpressure system. The leak rate of the overpressure system must be combined with the inner and outer seal leak rates and not exceed 1 x 10^{-5} ref cm^3/sec (1.0 x 10^{-5} mbar-l/sec). If the acceptance criterion is not met, locate the overpressure system leak, correct it, and retest.

11. Pressurize overpressure system (seal interspace) with helium to a pressure of about 6.0 atm abs (73.5 psig).

12. Install protective cover.

Note: The following step shall be performed in accordance with the timing set forth in the Technical Specifications.

TABLE 8.1-1 (continued)
SEQUENCE OF OPERATIONS

13. Verify that surface dose rates and surface contamination levels are within the limits set by the Technical Specifications. If the external dose rates above the neutron shield exceed the values in Technical Specification 5.2.3, install the shield ring (drawing 970-70-2, item 48) on the cask and re-measure the dose rates above the neutron shield. The user has the option to install this shield ring at an earlier stage in the loading operations, or to install the shield ring even if the measured dose rates without it are within the limits of Technical Specification 5.2.3.

    The neutron dose rate measuring instruments must be calibrated for a neutron energy spectrum appropriate to the exterior of the TN-68 cask.

Note: Steps 14 and 15 may be performed at the ISFSI if satisfactory lid and port cover seal testing is performed prior to moving the cask to the storage area.

14. Install pressure transducer/switch tubing on exterior of cask, and helium leak test to point of the valve at the protective cover. The total overpressure system leak rate combined with the inner and outer seal leak rates must be 1×10^{-5} ref cm^3c/sec (1.0×10^{-5} mbar-l/sec) or less. If the acceptance criterion is not met, locate the overpressure system leak, correct it and retest.

15. Backfill the external tubing with helium to a pressure of about 6.0 atm abs (73.5 psig) and open the valve at the protective cover.

16. Load cask on transporter.

17. Move cask to Storage Area.

<u>D. Storage Area</u>

1. Lower cask down onto storage pad in selected location. The cask spacing is controlled by the Technical Specifications.

2. Disconnect cask transporter.

3. Connect overpressure system to monitoring panel.

4. Perform Channel Operational Test (COT) to verify proper function of pressure switch/transducer.

TABLE 8.2-1
SEQUENCE OF OPERATIONS - UNLOADING

A. Storage Area

1. Disconnect overpressure system from monitoring panel.

2. Position cask transporter over cask.

3. Engage lifting arms and lift cask to designated lift height.

4. Move cask to spent fuel pool building.

B. Loading Area

1. Lower cask down onto floor, disconnect cask transporter and remove transporter.

2. Lift cask to decontamination area using lift beam and crane.

3. Remove neutron shield pressure relief valve and install plug in neutron shield vent hole.

4. Depressurize overpressure tank using the diaphragm valve, disconnect tubing at protective cover.

5. Remove protective cover.

6. Remove overpressure tank, overpressure port flange and top neutron shield.

7. Remove vent cover.

8. Collect a cavity gas sample through the vent port quick disconnect coupling.

9. Analyze the gas sample for radioactive material and add necessary precautions based on cavity gas sample results.

Note: If degraded fuel is suspected, additional measures, appropriate for the specific conditions, are to be planned, reviewed and approved by the designated approval authority, and implemented to minimize exposures to workers and radiological releases to the environment. These additional measures may include provision of filters, respiratory protection and other methods to control releases and exposures ALARA.

TABLE 8.2-1
SEQUENCE OF OPERATIONS - UNLOADING
(Continued)

10. In accordance with site requirements, vent cavity gas through the hose until atmospheric pressure is reached.

11. Remove vent port quick disconnect and drain port cover. Attach vent port adapter.

12. Loosen lid bolts and remove all but 6 approximately equally spaced lid bolts.

13. Attach cask to crane using lift beam. Attach lid lifting equipment.

14. Attach fill and drain lines to the drain quick disconnect coupling and the vent port adapter.

15. Ensure appropriate measures are in place to ensure proper handling of steam. Both fill and drain lines should be designed for steam at 100 psig minimum to prevent steam burns and radiation exposures due to line failure.

16. Lower cask into spent fuel pool/cask pit while spraying exterior of cask with demineralized water to minimize contamination. Lower until the cask top surface is just above the water level. Note: The cask may be filled before lowering the cask into the pool or with the cask partially submerged in the spent fuel pool.

C. Cask Loading Pool

Note: In BWR spent fuel pools, there may be significant amounts of fuel crud particulate material. Precautions should be taken to ensure that this particulate does not become airborne or become a radiation concern due to material floating on the surface of the water. Precautions may include enhanced filtering of the pool water during loading and unloading operations, increased ventilation and monitoring airborne contamination during all spent fuel pool activities.

1. Begin pumping pool or demineralized water into the cask through the drain port at a rate of no more than 3 gpm while continuously monitoring exit pressure (See Setup shown in Figure 8.2-1). Continue pumping at a rate up to 3 gpm for at least thirty five minutes. By this time, the water level in the cask will have reached the active fuel length.

2. The flow rate can then be gradually increased while monitoring the pressure at the outlet. If the pressure gage reading exceeds 65.6 psig, close the inlet valve until the pressure falls below 60 psig. Reflooding can then be resumed.

TABLE 8.2-1
SEQUENCE OF OPERATIONS - UNLOADING
(Continued)

3. Take a grab sample for chemistry analysis.

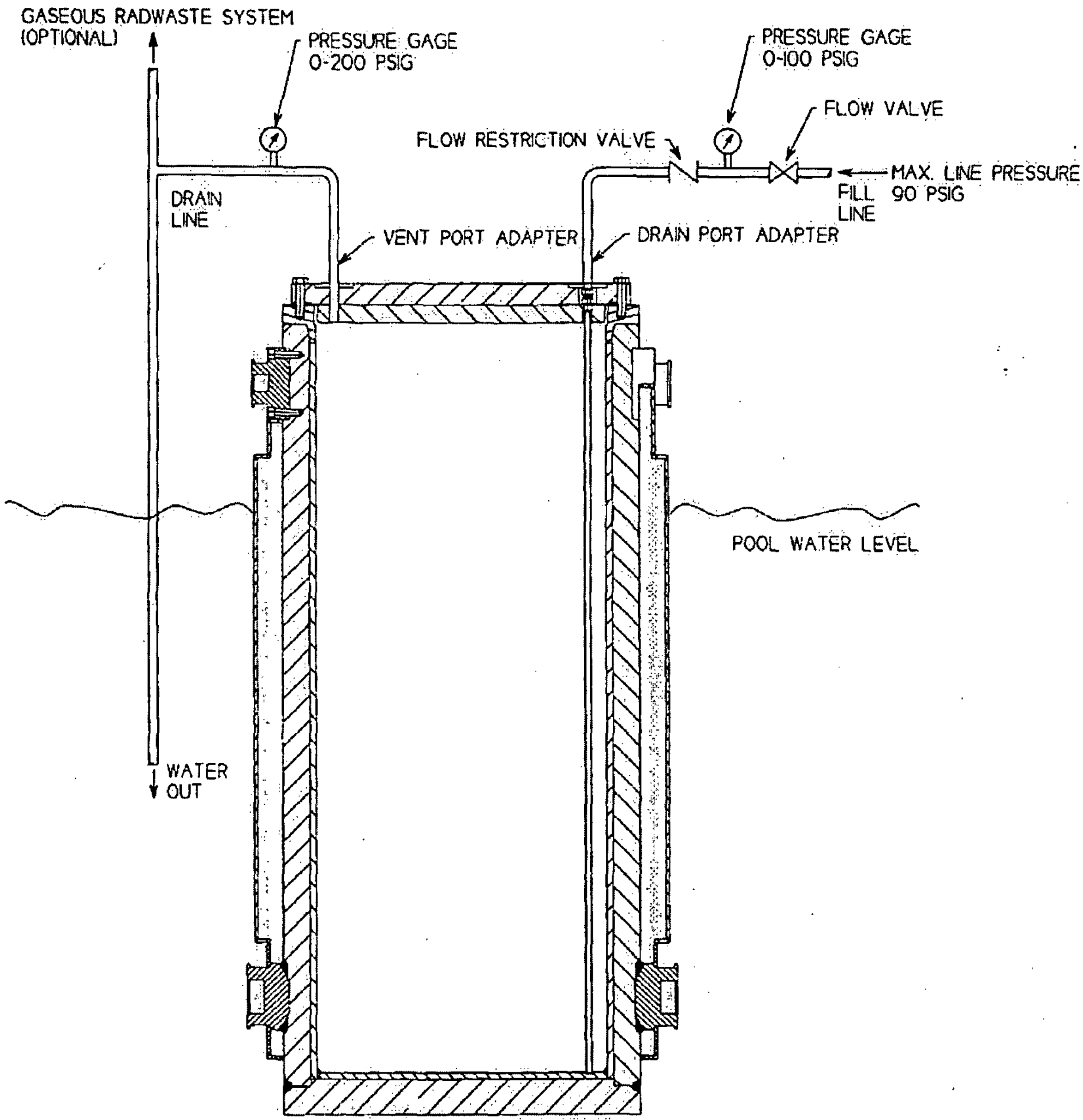
4. When the cask is full of water, remove the hose from the drain port and the hose and vent port adapter from the vent port. Remove the remaining 6 lid bolts.

5. Lower the cask and place it on the bottom of the pool/pit while rinsing the lift beam with demineralized water.

6. Raise the lift beam from the cask removing the cask lid.

7. Unload spent fuel assemblies in accordance with site procedures.

FIGURE 8.2-1.
TYPICAL SETUP FOR FILLING CASK WITH WATER



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CHAPTER 9

ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM

9.1 Acceptance Criteria

9.1.1 Visual Inspection

Visual inspections are performed at the Fabricator's facility to ensure that the casks conform to the drawings and specifications. The visual inspection includes verifying that all specified coatings are applied and the cask is clean and free of defects. (Visual inspection requirements on welds are discussed in Chapter 3.) Upon arrival at the loading facility, the casks are again inspected to ensure that the casks have not been damaged during shipment. Visual inspections which indicate conditions which are not in conformance with the drawings and specifications will be repaired or evaluated for the effect of the condition on the safety function of the components in accordance with 10CFR72.48 by the user.

9.1.2 Structural

The structural analyses performed on the cask are presented in Chapter 3. To ensure that the cask can perform its design function, all structural materials are chemically and physically tested to ensure that the required properties are met. All welding is performed using qualified processes and qualified personnel according to the ASME Boiler and Pressure Vessel Code(1). Base materials and welds are examined in accordance with ASME Boiler and Pressure Vessel code requirements. NDE requirements for welds are specified on the drawings provided in Chapter 1. All weld-related NDE is performed in accordance with written and approved procedures. Inspection personnel are qualified in accordance with SNT-TC-1A(2).

The confinement welds are designed, fabricated, tested and inspected in accordance with ASME B&PV Code Subsection NB. Exceptions to the code taken regarding the containment vessel are described in Chapter 7. The basket is designed, fabricated and inspected in accordance with the ASME B&PV Code, Section III, Subsection NG. Exceptions to the code taken regarding the basket are described in Section 3.1.2.3. The shield shell and lid shield plate are fabricated in accordance with ASME B&PV Code, Section III, Subsection NF except that post weld heat treatment of the bottom shield plate to shield shell weld is not required. Progressive examination of this weld is performed in accordance with Section 3E.1.2. Nonconfinement welds are inspected to the NDE acceptance criteria of ASME B&PV Code, Section III, Subsection NF.

A pressure test is performed on the cask assembly (containment vessel installed in gamma shield shell) at a test pressure of 125 psig, which is 1.25 times the design pressure of 100 psig. The test pressure is held for a minimum of 10 minutes. The test will be performed in accordance with ASME B&PV Code, Section III, Subsection NB, Paragraph NB-6200 or NB-6300. Visible joints/surfaces are visually examined for possible leakage after application of pressure. Temporary gaskets and seals may be used in place of the metallic seals during the test.

In addition, a bubble leak test is performed at 4.5 psig or greater on the resin enclosure. The purpose of this test is to identify any potential leak passages in the enclosure welds. The bubble leak test pressure is set at 1.5 times the relief valve set pressure.

The lifting trunnions are fabricated and tested in accordance with ANSI N14.6[(3)] and are designed for nonredundant (single failure proof) lifting. A load test of 3 times the design lift load is applied to the trunnions for a period of ten minutes to ensure that the trunnions can perform satisfactorily. The periodic load test or examination of the trunnions, including removal and inspection of the bolts in accordance with ANSI N14.6 will not be performed while the cask is in storage or prior to return of the cask from storage for unloading. This is justified since the cask will only be lifted a few times and there are no cyclic loads on the trunnions.

9.1.3 Leak Tests

Leakage tests are performed on the containment system and overpressure system at the Fabricator's facility. These tests are usually performed using the helium mass spectrometer method. Alternative methods are acceptable, provided that the required sensitivity is achieved. The leakage tests are performed in accordance with ANSI N14.5[(4)]. Personnel performing the leakage tests are qualified in accordance with SNT-TC-1A.

The containment boundary permissible leakage rate is less than or equal to 1×10^{-5} ref cm^3/sec. In order to assure the leakage rate of the containment boundary is less than 1×10^{-5} ref cm^3/sec, the total leak rate (of the inner seals and the outer seals) at standard conditions is less than 1×10^{-5} ref cm^3/sec. The sensitivity of the leakage test procedure is at least 5×10^{-6} ref cm^3/sec.

Although the overpressure system is not important to safety, it is also leak tested in accordance with ANSI N14.5. The permissible leakage rate for the overpressure system shall be less than or equal to 1×10^{-5} ref cm^3/sec. The sensitivity of the leakage test procedure shall be no less than 5×10^{-6} ref cm^3/sec.

9.1.4 Components

9.1.4.1 Valves

There are no valves performing a function important to safety. The TN-68 design incorporates quick-connect couplings for ease of draining and venting. However, these couplings do not form part of the containment boundary. They are covered by bolted closures with metallic o-ring seals.

9.1.4.2 Gaskets

The lid and all containment penetrations are sealed using double metallic o-ring seals. The inside o-ring forms part of the containment boundary. Metallic o-rings are not temperature sensitive, and are therefore tested at room temperature. Metallic o-rings of the same type as

those to be used for storage are installed for the fabrication leakage test described in Section 9.1.3. The tested o-rings are replaced before loading the cask. Upon completion of cask loading, the seals are tested in accordance with Technical Specifications Surveillance Requirement 3.1.3.1.

9.1.5 Shielding Integrity

The analyses performed to ensure shielding integrity are presented in Chapter 5.
The radial neutron shield is protected from damage or loss by the aluminum and steel enclosure. The material is a proprietary borated reinforced polymer.

The resin's primary function is neutron shielding, which is provided primarily by its hydrogen content. The resin includes boron to reduce secondary gamma radiation which occurs when neutrons are captured by hydrogen. Both neutrons and capture gammas are a small component of the radial dose rate. The resin also provides some gamma shielding, which is a function of the overall resin density, but is not sensitive to composition.

The shielding performance of the material can be adequately verified by chemical analysis and verification of density. Uniformity is assured by installation process control.

The following are acceptance values for density and chemical composition for the resin. The values used in the shielding calculations of Chapter 5 are included for comparison.

Chapter 5 values		Acceptance Testing Values		
Element	nominal wt %	Element	wt %	acceptance range (wt %)
H	5.05	H	5.05	-10 / +20
B	1.05	B	1.05	± 20
C	35.13	C	35.13	± 20
Al	14.93	Al	14.93	± 20
O	41.73	O+Zn (balance)	43.84	± 20
Zn				
Total	97.89%		100%	

The nominal resin density used in Chapter 5 calculations is 1.58 g/cm^3. However, because zinc is not included, the sum of the individual elements is only 97.89%, and the effective density used in the shielding calculations is 1.58*0.9789 = 1.547 g/cm^3. Therefore, the minimum resin density in acceptance testing is 1.547 g/cm^3.

Density testing will be performed on every mixed batch of resin. Chemical analysis will be made on the first batch mixed with a given set of components, and thereafter whenever a new lot of one of the major components is introduced. Major components are aluminum oxide, zinc borate and the polyester resin, which combined make up 92% of the resin by weight.

Qualification tests of the personnel and procedure used for mixing and pouring the polyester resin used for radial neutron shielding are performed. Qualification testing includes verification that the chemical composition and density is achieved, and the process is performed in such a manner as to prevent voids. Tests are performed at loading to ensure that the radiation dose limits are not exceeded for each cask.

9.1.6 Thermal Acceptance

The heat transfer analysis for the basket includes credit for the thermal conductivity of neutron-absorbing materials, as specified in Section 4.2 part 12. Because these materials do not have publicly documented values for thermal conductivity, testing of such materials will be performed in accordance with Section 9.4.1.

9.1.7 Neutron Absorber Tests

CAUTION

Sections 9.1.7.1 through 9.1.7.3 below are incorporated by reference into the TN-68 CoC 1027 Technical Specifications (paragraph 4.1.1) and shall not be deleted or altered in any way without a CoC amendment approval from the NRC. The text of these sections is shown in bold type to distinguish it from other sections.

The neutron absorber used for criticality control in the TN-68 basket may consist any of the following types of material:

(a) Boron-aluminum alloy (borated aluminum)

(b) Boron carbide / aluminum metal matrix composite (MMC)

(c) Boral®

The boron content of these materials is given by Table 9.1-1.

The neutron absorber plates may be monolithic, or they may consist of paired plates, one containing boron in the specified areal density, and the other composed of aluminum or aluminum alloy to make up the balance of the specified thickness and thermal conductance.

The TN-68 safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials. To assure performance of the neutron absorber's design function the presence of B10 and the uniformity of its distribution need to be verified by acceptance testing as specified in Section 9.4.2, with the exception of the materials for units TN-68-1 through 44, which may be accepted by the testing described in Section 9.5.

9.1.7.1 Boron Aluminum Alloy (Borated Aluminum)

See the Caution in Section 9.1.7 before deletion or modification to this section.

The material is produced by direct chill (DC) or permanent mold casting with boron occurring as a uniform fine dispersion of discrete AlB_2 or TiB_2 particles in the matrix of aluminum or aluminum alloy. For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, either the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product, with sufficient margin to minimize rejection, typically 10 % excess. The amount required to achieve the specified minimum B10 areal density will depend on whether boron with the natural isotopic distribution of the isotopes B10 and B11, or boron enriched in B10 is used. In no case shall the boron content in the aluminum or aluminum alloy exceed 5% by weight.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section 9.4.2 or 9.5. The specified acceptance testing assures that at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products and Castings."[5] Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable. Widespread blisters, rough surface, or cracking shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

9.1.7.2 Boron Carbide / Aluminum Metal Matrix Composites (MMC)

See the Caution in Section 9.1.7 before deletion or modification to this section.

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, or thermal spray techniques. It is a low-porosity product, with a metallurgically bonded matrix. The boron carbide content shall not exceed 40% by volume.

Prior to use in the TN-68, MMCs shall pass the qualification testing specified in Section 9.4.3, and shall subsequently be subject to the process controls specified in Section 9.4.4.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section 9.4.2. The specified acceptance testing assures that at any location in the final product, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products and Castings."[5]

Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable. Widespread blisters, rough surfaces, or cracking shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

References to metal matrix composites throughout this chapter are not intended to refer to Boral®, which is described in the following section.

9.1.7.3 Boral®

See the Caution in Section 9.1.7 before deletion or modification to this section.

This material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum, mechanically bonded by hot-rolling an "ingot" consisting of an aluminum box filled with blended boron carbide and aluminum powders. The core, which is exposed at the edges of the sheet, is slightly porous. The average size of the boron carbide particles is approximately 80 microns before and somewhat smaller after rolling. The nominal boron carbide content shall be limited to 65% (+ 2% tolerance limit) of the core by weight.

The criticality calculations take credit for 75% of the minimum specified B10 areal density of Boral®. B10 areal density will be verified by chemical analysis and by certification of the B10 isotopic fraction for the boron carbide powder, or by neutron transmission testing. Areal density testing is performed on an approximately 1 cm^2 area from the thinnest coupon, typically that taken near one of the corners of the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be treated as non-conforming. Alternatively, individual pieces cut from the sheet may be accepted if a coupon from the sheet, thinner than any location on the piece in question, has a measured areal density equal to or greater than that specified.

Visual inspections shall verify that the Boral® core is not exposed through the face of the sheet at any location.

9.2 Maintenance Program

Because of their passive nature, the storage casks will require little, if any, maintenance over the lifetime of the ISFSI. Typical maintenance tasks could involve occasional recalibration of pressure monitoring instrumentation and repainting of some casks with corrosion-inhibiting coatings. No special maintenance techniques are necessary.

Two identical pressure transducers/switches are provided. If the instrument malfunctions, the second switch can be connected. The pressure transducers/switches are not replaced unless they are malfunctioning.

All the gaskets used for the containment boundary are metallic o-rings. They are designed to maintain their sealing capability until the cask is reopened. If a leak is detected by a drop in pressure in the overpressure system, all the gaskets can be replaced. For a drop in pressure that is consistent with the maximum allowable leak rate (see Figure 7.1-1), the overpressure system can be re-pressurized at the time of transducer/switch maintenance.

9.3 Marking

The TN-68 is marked with the model number, unique identification number, and empty weight in accordance with 10 CFR 72.236(k). The unit identification number has the form TN-68-XX-Y-Z, where XX is a sequential number corresponding to a specific cask, Y is blank or a letter from A to G indicating B10 areal density in the basket neutron absorber plates (see Table 9.1-1) and Z is blank or the letter Q indicating that the basket is outfitted to accommodate damaged fuel.

9.4 Specification for Neutron Absorbers

9.4.1 Specification for Thermal Conductivity Testing of Neutron Absorbers

Testing shall conform to ASTM E1225[7], ASTM E1461[8], or equivalent method, performed at room temperature on coupons taken from the rolled or extruded production material. Previous testing of borated aluminum and metal matrix composite, Table 9.4-1, shows that thermal conductivity increases slightly with temperature. Initial sampling shall be one test per lot, defined by the heat or ingot, and may be reduced if the first five tests meet the specified minimum thermal conductivity.

If a thermal conductivity test result is below the specified minimum, additional tests may be performed on the material from that lot. If the mean value of those tests falls below the specified minimum (Ch 4, Section 4.2, item 12), the associated lot shall be rejected.

After twenty five tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the boron appearing in the same phase, e.g., B_4C, TiB_2, or AlB_2, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.

The thermal analysis in Chapter 4 considers a base model with 0.31" thick neutron absorber. This model gives the bounding values for the maximum component temperatures. The dual plate basket construction alternate model described in Section 4.3.1 assumes a 3/16 inch thick neutron absorber paired with a 1/8 inch thick aluminum 1100 plate to make a total thickness of 0.31". The specified thickness of the neutron absorber may vary, and the thermal conductivity acceptance criterion for the neutron absorber will be based on the nominal thickness specified. To maintain the thermal performance of the basket, the minimum thermal conductivity shall be such that the total thermal conductance (sum of conductivity * thickness) of the neutron absorber and the aluminum 1100 plate shall equal the conductance assumed in the analysis for the base model. Samples of the acceptance criteria for various neutron absorber thicknesses are highlighted in Table 9.4-2.

The aluminum 1100 plate does not need to be tested for thermal conductivity; the material may be credited with the values published in the ASME Code Section II part D.

9.4.2 Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission

CAUTION
Section 9.4.2 is incorporated by reference into the TN-68 CoC 1027 Technical Specifications (paragraph 4.1.1) and shall not be deleted or altered in any way without a CoC amendment approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

For TN-68 units 01 through 44, Neutron Transmission testing is performed per Section 9.5 of this chapter.

Neutron Transmission acceptance testing procedures shall be subject to approval by the Certificate Holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam of up to 1.2 centimeter diameter. A beam size greater than 1.2 centimeter diameter but no larger than 1.7 centimeter diameter may be used if computations are performed to demonstrate that the calculated $k_{effective}$ of the system is still below the calculated Upper Subcritical Limit (USL) of the system assuming defect areas the same area as the beam.

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard.

Alternatively, digital image analysis may be used to compare neutron radioscopic images of the test coupon to images of the standards. The area of image analysis shall be up to 1.1 cm^2.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. The following illustrates one acceptable method.

The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The minimum B10 areal densities determined by neutron transmission are converted to volume density, i.e., the minimum B10 areal density is divided by the thickness at the location of the neutron transmission measurement or the maximum thickness of the coupon. The lower tolerance limit of B10 volume density is then determined, defined as the mean value of B10 volume density for the sample, less K times

the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence[16]. If a goodness-of-fit test demonstrates that the sample comes from a normal population, the value of K for a normal distribution may be used. Otherwise, use a non-parametric (distribution-free) method of determining the one-sided tolerance limit.

Finally, the minimum specified value of B10 areal density is divided by the lower tolerance limit of B10 volume density to arrive at the minimum plate thickness which provides the specified B10 areal density.

Any plate which is thinner than this minimum or the minimum design thickness, whichever is greater, shall be treated as non-conforming, with the following exception. Local depressions are acceptable, so long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness.

Non-conforming material shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

9.4.3 Specification for Qualification Testing of Metal Matrix Composites

9.4.3.1 Applicability and Scope

Metal matrix composites (MMCs) shall consist of fine boron carbide particles in an aluminum or aluminum alloy matrix. The ingot shall be produced by either powder metallurgy (PM), thermal spray techniques, or by direct chill (DC) or permanent mold casting. In any case, the final MMC product shall have density greater than 98% of theoretical, a metallurgically bonded matrix, and boron carbide content no greater than 40% by volume. Boron carbide particles for the products considered here typically have an average size in the range 10-40 microns, although the actual specification may be by mesh size, rather than by average particle size. No more than 10% of the particles shall be over 60 microns. The material shall have negligible interconnected porosity exposed at the surface or edges.

Prior to initial use in a spent fuel dry storage or transport system, such MMCs shall be subjected to qualification testing that will verify that the product satisfies the design function. Key process controls shall be identified per Section 9.4.4 so that the production material is equivalent to or better than the qualification test material. Changes to key processes shall be subject to qualification before use of such material in a spent fuel dry storage or transport system.

ASTM test methods and practices are referenced below for guidance. Alternative methods may be used with the approval of the certificate holder.

9.4.3.2 Design Requirements

In order to perform its design functions the product must have at a minimum sufficient strength and ductility for manufacturing and for the normal and accident conditions of the storage/

transport system. This is demonstrated by the tests in Section 9.4.3.4. It must have a uniform distribution of boron carbide. This is demonstrated by the tests in Section 9.4.3.5.

9.4.3.3 Durability

There is no need to include accelerated radiation damage testing in the qualification. Such testing has already been performed on MMCs, and the results confirm what would be expected of materials that fall within the limits of applicability cited above. Metals and ceramics do not experience measurable changes in mechanical properties due to fast neutron fluences typical over the lifetime of spent fuel storage, about 10^{15} neutrons/cm^2.

The need for thermal and corrosion (hydrogen generation) testing shall be evaluated case-by-case based on comparison of the material composition and environmental conditions with previous thermal or corrosion testing of MMCs.

Thermal testing is not required for MMCs consisting only of boron carbide in an aluminum 1100 matrix, because there is no reaction between aluminum and boron carbide below 842 °F[9], well above the basket temperature under normal conditions of storage or transport.

Corrosion testing is not required for full density MMCs consisting only of boron carbide in an aluminum 1100 matrix, because testing on one such material has already been performed by Transnuclear[15].

9.4.3.4 Required Qualification Tests and Examinations to Demonstrate Mechanical Integrity

At least three samples, one each from the two ends and middle of the test material production run shall be subject to:

a) room temperature tensile testing (ASTM- B557[10]) demonstrating that the material has the following tensile properties:

   - Minimum yield strength, 0.2% offset: 1.5 ksi
   - Minimum ultimate strength: 5 ksi
   - Minimum elongation in 2 inches: 0.5%
     (Alternatively show that the material fails in a ductile manner, e.g., by scanning electron microscopy of the fracture surface or by bend testing.)

and

b) testing (ASTM-B311[11]) to verify more than 98% of theoretical density. Testing or examination for exposed interconnected porosity shall be performed by a means to be approved by the Certificate Holder.

9.4.3.5 Required Tests and Examinations to Demonstrate B10 Uniformity

CAUTION *Section 9.4.3.5 is incorporated by reference into the TN-68 CoC 1027 Technical Specifications (paragraph 4.1.1) and shall not be deleted or altered in any way without a CoC amendment approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.*

Uniformity of the boron distribution shall be verified either by:

(a) Neutron radioscopy or radiography (ASTM E94[12], E142[13], and E545[14]) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or

(b) Quantitative testing for the B10 areal density, B10 density, or the boron carbide weight fraction, on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section 9.4.2, or by chemical analysis for boron carbide content in the composite.

9.4.3.6 Testing for Other Design Properties

If the design depends upon the thermal conductivity of the material, at least one specimen from the test material shall be subject to thermal conductivity testing (ASTM E1225[7] or ASTM E1461[8]) to verify that the material has the specified minimum thermal conductivity at all temperatures specified in the design.

9.4.3.7 Approval of Procedures

Qualification procedures shall be subject to approval by the Certificate Holder.

9.4.4 Specification for Process Controls for Metal Matrix Composites

9.4.4.1 Applicability and Scope

The applicability of this section is the same as that of Section 9.4.3. This section addresses the process controls to ensure that the material delivered for use is equivalent to the qualification test material.

Key processing changes shall be subject to a complete program of qualification testing per Section 9.4.3 prior to use of the material produced by the revised process. The Certificate Holder shall determine whether a complete or partial re-qualification program per Section 9.4.3 is required, depending on the characteristics of the material that could be affected by the process change.

9.4.4.2 Definition of Key Process Changes

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduce density, or reduce the mechanical strength or ductility of the MMC.

9.4.4.3 Identification and Control of Key Process Changes

> *CAUTION*
> *Section 9.4.4.3 is incorporated by reference into the TN-68 CoC 1027 Technical Specifications (paragraph 4.1.1) and shall not be deleted or altered in any way without a CoC amendment approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.*

The manufacturer shall provide the Certificate Holder with a description of materials and process controls used in producing the MMC. The Certificate Holder and manufacturer shall identify key process changes as defined in Section 9.4.4.2.

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change. The following are examples of other changes that may be established as key process changes, as determined by the Certificate Holder's review of the specific applications and production processes:

(a) Changes in the boron carbide particle size specification that increase the average particle size by more than 5 microns or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit,

(b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering,

(c) Change in the nominal matrix alloy,

(d) Changes in mechanical processing that could result in reduced density of the final product, e.g., for PM or thermal spray MMCs that were qualified with extruded material, a change to direct rolling from the billet,

(e) For MMCs using a 6000 series aluminum matrix, changes in the billet formation process that could increase the likelihood of magnesium reaction with the boron carbide, such as an increase in the maximum temperature or time at maximum temperature, and

(f) Changes in powder blending or melt stirring processes that could result in less uniform distribution of boron carbide, e.g., change in duration of powder blending.

In no case shall process changes be accepted if they result in a product outside the limits in Sections 9.4.3.1 and 9.4.3.4.

9.5 Alternate Acceptance Testing for Neutron Absorbers on TN-68-01 through -44

Neutron absorber material for the first forty-four TN-68 casks consisted either of borated aluminum (1.7% boron, minimum 30 mg B10/cm^2), or Boralyn® MMC (15% B_4C, minimum 36 mg B10/cm^2). These materials were manufactured prior to October 2004, and were subject to the original TN-68 neutronic acceptance testing described here.

9.5.1 Test Coupons

Each neutron absorber plate is 10.4 inches wide by ~42, 55, and 69 inches long. Coupons the full width of the plate (10.4 inches) will be removed between each finished plate and at the ends of the "stock plate". The thermal conductivity coupon may be removed from one of the neutronic inspection coupons. The minimum dimension of the coupon shall be as required for neutron transmission measurements; 1 to 2 inches is adequate for the typical 1 cm diameter neutron beam.

9.5.2 Acceptance Testing

Effective boron 10 content is verified by neutron transmission testing of these coupons. The transmission through the coupons is compared with transmission through calibrated standards composed of a homogeneous boron compound without other significant neutron absorbers, for example zirconium diboride or titanium diboride. These standards are paired with aluminum shims sized to match the scattering by aluminum in the neutron absorber plates. Provision shall be made so that the neutron transmission test is not always made in the same location on the coupon. Thus, the random placement of the coupons in the test fixture results in testing at two locations across the plate width. The effective B10 content of each coupon, minus 3σ based on the number of neutrons counted for that coupon, must be greater than the specified areal density. Rejection of a given coupon shall result in rejection of the contiguous plate(s).

Macroscopic uniformity of B10 distribution is verified by neutron radioscopy of the coupons. The acceptance criterion is that there be uniform luminance across the coupon. This inspection shall cover the entire coupon.

Normal sampling of coupons for neutron transmission measurements and radioscopy shall be 100%. Reduced sampling (50%) may be introduced based upon acceptance of all coupons in the first 25% of the lot. A rejection during reduced inspection will require a return to 100% inspection of the lot. A lot is defined as all the plates produced from a single casting or powder metal billet.

9.6 References

1. ASME Boiler and Pressure Vessel Code, Section III, 1995 Edition including 1996 addenda.

2. SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 1992.

3. ANSI N14.6, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials," New York, 1986.

4. ANSI N14.5-1997, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials", February 1998.

5. "Aluminum Standards and Data, 2003" The Aluminum Association.

6. "Topical Report: Credit for 90% of the ^{10}B in BORAL," AAR Report 1829, AAR Manufacturing, Oct 2004

7. ASTM E1225, "Thermal Conductivity of Solids by Means of the Guarded-Comparative-Longitudinal Heat Flow Technique"

8. ASTM E1461, "Thermal Diffusivity of Solids by the Flash Method"

9. Pyzak and Beaman, "Al-B-C Phase Development and Effects on Mechanical Properties of B_4C/Al Derived Composites," J. Am. Ceramic Soc., 78[2], 302-312 (1995)

10. ASTM B557, "Standard Test Methods of Tension Testing Wrought and Cast Aluminum- and Magnesium-Alloy Products"

11. ASTM B311, "Test Method for Density Determination for Powder Metallurgy (P/M) Materials Containing Less Than Two Percent Porosity"

12. ASTM E94, "Recommended Practice for Radiographic Testing"

13. ASTM E142, "Controlling Quality of Radiographic Testing"

14. ASTM E545, "Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing"

15. "Hydrogen Generation Analysis Report for TN-68 Cask Materials," Test Report No. 61123-99N, Rev 0, Oct 23, 1998, National Technical Systems

16. Natrella, "Experimental Statistics," Dover, 2005

Table 9.1-1
Boron Content of Neutron Absorbers

Borated Aluminum & MMCs, 90% B10 Credit

Basket Designator "Y"	Specified minimum areal density g B10/cm^2	Maximum fuel enrichment (note 1)	Nom wt % boron in enriched borated aluminum 0.3 inch thick (notes 2, 3)	Nominal vol % B_4C in MMC, 0.3 inch thick (notes 2, 3)
none	30	3.70	1.55	11.0
A	35	3.95	1.80	12.9
B	40	4.05	2.06	14.7
C	45	4.15	2.32	16.5
D	50	4.30	2.58	18.4
E	55	4.40	2.84	20.2
F	60	4.50	3.09	22.1
G	70	4.70	3.61	25.8

Boral®, 75% B10 Credit

Basket Designator "Y"	Specified minimum areal density g B10/cm^2	Maximum fuel enrichment (note 1)	Boral® nominal core thickness, inch (note 2)
none	36	3.70	0.077
A	42	3.95	0.088
B	48	4.05	Note 4
C	54	4.15	
D - G	60 - 84	4.30 – 4.70	

Notes:

1. Lattice average enrichment limit for undamaged fuel, pellet enrichment limit for damaged fuel
2. The neutron absorber manufacturer may adjust the amount of boron as required to achieve the specified minimum areal density.
3. If a neutron absorber thinner than 0.3 inch is paired with an aluminum plate, the boron content varies in inverse proportion to the thickness to maintain the same areal density
4. Use of Boral® in this range is not anticipated due to thermal conductivity limitations

Table 9.4-1
Thermal Conductivity as a Function of Temperature for Sample Neutron Absorbers

Temperature °C	Material 1	Material 2	Material 3	Material 4
20	193	170	194	194
100	203	183	207	201
200	208	-	-	
250	-	201	218	206
300	211	204	220	203
314	-	-	-	202
342	-	-	-	202

Units: W/mK

Materials:
1) Boralyn® MMC, aluminum 1100 with 15% B_4C
2) Borated aluminum 1100, 2.5% boron as TiB_2
3) Borated aluminum 1100, 2.0% boron as TiB_2
4) Borated aluminum 1100, 4.3% boron as AlB_2

Sources:

Thermal Conductivity Measurements of Boron Carbide/Aluminum Specimens, Oct 1998, testing by Precision Measurements and Instruments Corp. for Transnuclear, Inc.

Qualification of Thermal Conductivity, Borated Aluminum 1100, Eagle Picher Report AAQR06, May 2001

Table 9.4-2
Sample Determination of Thermal Conductivity Acceptance Criterion

Base Model	Al 1100	n absorber	total	
thickness (inch)	0	0.31	0.31	as modeled
conductivity at 70°F (Btu/hr-in-°F)	n/a	**7.94**	n/a	
conductance (Btu/hr-°F)	0	2.46	2.46	

Dual Plate Construction	Al 1100	n absorber	total	
thickness (inch)	0.1225	0.1875	0.31	as modeled
conductivity at 70°F (Btu/h-.in-°F)	11.09	**7.94**	n/a	
conductance (Btu/hr-°F)	1.36	1.49	2.85	

thickness (inch)	0.06	0.25	0.31	thicker neutron absorber
conductivity at 70°F (Btu/hr-in-°F)	11.09	**8.72**	n/a	
conductance (Btu/hr-°F)	0.67	2.18	2.85	

thickness (inch)	0.185	0.125	0.31	thinner neutron absorber
conductivity at 70°F (Btu/hr-in-°F)	11.09	**6.40**	n/a	
conductance (Btu/hr-°F)	2.05	0.80	2.85	

The acceptance criterion is identified by boldface type for each thickness.

The neutron absorber material need not be tested for thermal conductivity if the nominal thickness of the aluminum 1100 in the paired plates is 0.237 inch or greater. The conductance of such plate is equal to 2.46 Btu/hr-°F at the lowest conductivity for Al-1100 (10.4 Btu/hr-in-°F @ 400°F) and satisfies the above criteria for the base model.

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APPENDIX 9A

Transnuclear TN-68 Radial Neutron Shield Material

The material is an unsaturated polyester crosslinked with styrene, with about 50 weight % mineral and fiberglass reinforcement. The components are polyester resin, styrene monomer, alpha methyl styrene, aluminum oxide, zinc borate, and chopped fiberglass.

Thermal Stability

Thermal aging tests on a material with the same components in slightly different proportions have been performed by Transnucleaire, Paris (TNP). The tests by TNP evaluate weight loss and offgassing at 125 °C (260 °F) and 155 °C (311 °F). The maximum normal temperature in the TN-68 radial neutron shield is 259 °F (126 °C) at the beginning of storage per Chapter 4 of the TN-68 SAR. An exponential weight loss occurs that rapidly approaches a maximum value. After 106 hours, the weight loss is about 1.0%, and extrapolation of the results indicates maximum weight loss of about 1.3%. This effect diminishes rapidly with decreasing temperature, as can be seen by comparing the results at 125 and 155 °C in Figure 9A-1. An analysis of the gas released from a sample heated from 25 to 125 °C over one hour shows it to be 99.9% styrene. The results are included in the attached Table 9A-1.

These results obtained with small samples (50 mm thick x 50 mm dia) are conservative with respect to the material in a larger enclosed form such as the TN-68 radial neutron shield, where volatile constituents must diffuse through a much greater distance to be released.

Radiation Stability

The European Organization for Nuclear Research (CERN) has published a compilation of its own testing and of prior published data on the radiation resistance of various materials. Volume Two[1] presents the results of testing, and Volume Three[2] summarizes the results and provides recommendations in Appendices 5.9 and 6. These show that while unfilled polyester has poor radiation resistance, both mineral- and glass-filled polyester, such as used in

the TN-68 radial neutron shield, are among the most radiation-resistant of thermosetting resins.

References

1. Schönbacher, et. al. , CERN 79-08, Compilation of Radiation Damage Test Data, Part II, Thermosetting and thermoplastic resins, 15 Aug 1979
2. Beynel, et. al., CERN 82-10, Compilation of Radiation Damage Test Data, Part III, Materials used around high-energy accelerators, 4 Nov 1982

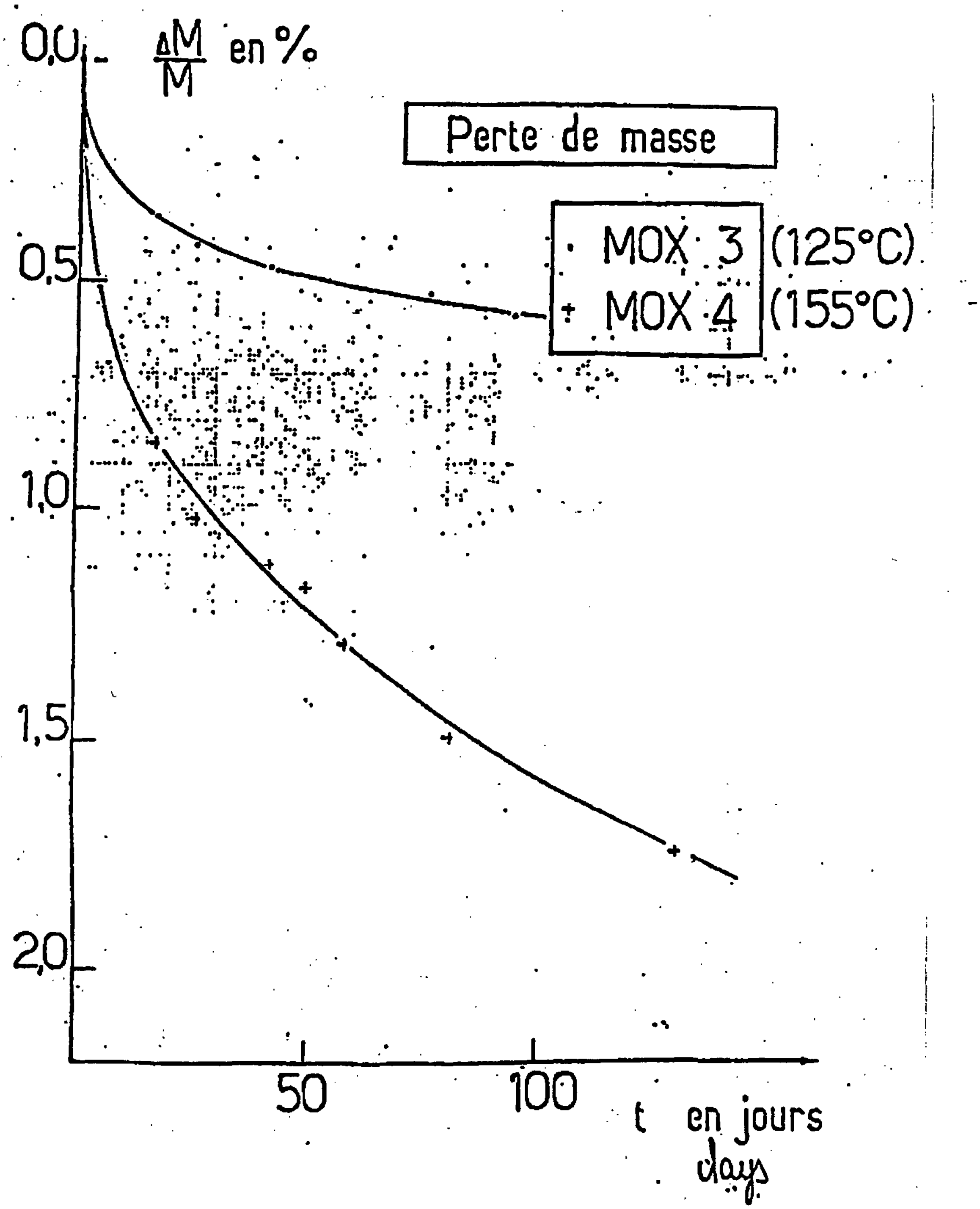
Table 9A-1

Quantitative Analysis of Gases Released from Neutron Shield Test Resin Upon Heating from 21 to 125 °C for One Hour

Gas analyzed	Quantity (µg/g of resin)
Styrene	610
O_2	0.030
N_2	0.21
H_2	0.005
CO	0.03
CO_2	0.24
CH_4	<0.0005
C_2H_4	<0.001
C_2H_5	<0.001
C_3H_6	<0.003
C_3H_8	<0.003
iso-C_4H_{10}	<0.006
n-C_4H_{10}	<0.006
iso-C_5H_{10}	<0.02
n-C_5H_{10}	<0.02

Figure 9A-1

Weight Loss Due to Thermal Aging of Neutron Shield Test Resin



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CHAPTER 10

RADIATION PROTECTION

10.1 Ensuring that Occupational Radiation Exposures are As Low As Is Reasonably Achievable (ALARA)

10.1.1 Policy Considerations

A radiological protection program will be implemented at the ISFSI in accordance with requirements of 10CFR72.126. The program will be based upon the specific policies in existence at the nuclear generating plant, (ISFSI license holder).

Plant personnel are given training in the proper operation of the cask. This training covers operations, inspections, repair and maintenance of the cask. Proper training of the operation personnel helps to minimize exposure to radiation such that the total individual and collective exposure to personnel in all phases of operation and maintenance are kept As Low As Reasonably Achievable.

10.1.2 Design Considerations

The TN-68 cask is designed to store BWR fuel assemblies. It is similar in design to the TN-32 and TN-40 in use at Surry and Prairie Island, respectively, which store PWR assemblies. Experience from these sites has shown the TN designs have good operational features that have resulted in occupational exposures being lower than those postulated here.

The TN-68 dry storage cask design takes into account radiation protection considerations, which ensure that occupational radiation exposures are ALARA. The fuel will be stored dry, inside the sealed, heavily-shielded cask. The most significant radiation protection design consideration provides for heavy shielding to minimize personnel exposures. To avoid personnel exposures, the casks will not be opened nor fuel removed from the casks while at the ISFSI. Storage of the fuel in the dry sealed cask eliminates the possibility of leakage of contaminated liquids. Gaseous releases are not considered credible. The exterior of the casks will be decontaminated before leaving the station, thereby minimizing exposure of personnel to surface contamination and the potential spread of contamination outside of the radiologically controlled area. The TN-68 cask contains no active components which require periodic maintenance or surveillance. This method of spent fuel storage minimizes direct radiation exposures and eliminates the potential for personnel contamination.

Regulatory Position 2 of Regulatory Guide 8.8[1], is incorporated into the design considerations, as described below:

- ALARA objective 2a on access control will be met by use of a fence with a locked gate that surrounds the ISFSI and prevents unauthorized access.

- Regulatory Position 2b on radiation shielding is met by the heavy shielding of the cask which minimizes personnel exposures.

- Regulatory Position 2c on process instrumentation and controls is met by designing the instrumentation for a long service life and locating readouts in a low dose rate location.

- Regulatory Position 2d on control of airborne contaminants does not apply because no gaseous releases are expected. No significant surface contamination is credible because the exterior of the cask will be decontaminated before it leaves the station.

- Regulatory Position 2e on crud control is not applicable to the ISFSI because there are no systems at the ISFSI that could transport crud.

- Regulatory Position 2f on decontamination is met because the exterior of the cask is designed for decontamination. The cask is decontaminated before it is released from the decontamination areas in the station.

- Regulatory Position 2g on radiation monitoring does not apply because the casks are sealed. There is no need for airborne radioactivity monitoring since no airborne radioactivity is anticipated. Area radiation monitors are not required because the ISFSI will not normally be occupied.

- Regulatory Position 2h on resin treatment systems is not applicable because there will be no radioactive systems containing resins.

- Regulatory Position 2i concerning other miscellaneous ALARA items is not applicable because these items refer to radioactive systems not present at the ISFSI.

10.1.3 Operational Considerations

Operational requirements for surveillance are incorporated into the design considerations in Section 10.1.2 in that the casks are stored with adequate spacing to allow ease of on site surveillance. In addition, remote annunciation and/or indication is available outside the ISFSI protected area to minimize surveillance time.

The operational requirements are incorporated into the radiation protection design features described in Section 10.2 since the cask are heavily shielded to minimize occupational exposure.

The TN-68 cask is designed to be essentially maintenance free. It is a passive system without any moving parts. The double metallic O-ring design with periodic surveillance of the over pressure system guarantees that in the unlikely event of a failure of one of the seals, adequate time is available to restore the cask leak tightness.

The only cask repair procedure that could be envisioned is the replacement of a containment seal. For this, the cask would be returned to the spent fuel pool area in order to minimize radiation exposure to personnel.

The only anticipated maintenance procedures are visual inspection, possible paint touch-up, pressure transducer/switch surveillance/maintenance, and overpressure system re-pressurization.

The TN-68 cask/ISFSI contains no systems that process liquids or gases or contain, collect, store, or transport radioactive liquids or solids other than the stored spent fuel. Therefore, the ISFSI meets ALARA requirements since there are no such systems to be maintained, be repaired, or be a source of leaks.

10.2 Radiation Protection Design Features

10.2.1 Cask Design Features

The TN-68 dry storage cask has a number of design features which ensure a high degree of integrity for the confinement of radioactive materials and reduction of direct radiation exposures to levels that are as low as practical.

- The casks are loaded, sealed, and decontaminated prior to transfer to the ISFSI.

- The fuel will not be unloaded nor will the casks be opened at the ISFSI.

- The fuel will be stored dry inside the casks, so that no radioactive liquid is available for leakage.

- The casks will be sealed airtight with a helium atmosphere to preclude oxidation of the fuel. The seals are double metallic o-rings to assure leak-tightness.

- The casks will be heavily shielded to reduce external dose rates. The shielding design features are discussed below.

- No radioactive material will be discharged during storage.

Shielding for the TN-68 cask is provided mainly by the thick-walled cask body. For neutron shielding, a borated polyester resin compound surrounds the cask body and a polypropylene disk covers the lid for storage. Additional shielding is provided by the steel shell surrounding the resin layer and by the stainless steel and aluminum/steel structure of the basket. Details of the cask shielding and radioactive sources are provided in Section 5.2.

Geometric attenuation, enhanced by ground and air attenuation, provides additional "shielding" for distant locations at restricted area and site boundaries. Two independent methods are utilized to determine the dose rates (and annual doses) around a 20-cask ISFSI containing design basis fuel.

10.2.2 Radiation Dose Rates

Calculated dose rates in the immediate vicinity of the TN-68 cask are presented in Chapter 5, which provides a detailed description of source term configuration, analysis models and bounding dose rates. The dose rate as a function of distance from the ISFSI for a single TN-68 cask is also presented in Chapter 5. Off-site dose rates and annual doses for an ISFSI containing 20 casks are presented in this section. This evaluation determines the neutron and gamma-ray off-site dose rates including skyshine in the vicinity of a generic ISFSI layout containing design-basis contents in the casks.

The generic ISFSI evaluated is a 2x10 array of TN-68 casks loaded with design-basis (described in chapter 5 as DBF-68) fuel. This generic ISFSI layout is shown in Figure 10.2-1. This

evaluation provides results for distances ranging from 10 to 600 meters from the front, side and corner of the arrays.

The total annual exposure for this ISFSI layout as a function of distance from each location (front, corner or side) is given in Table 10.2-1 and plotted in Figure 10.6. The total annual exposure estimates assume 100% occupancy for 365 days.

The Monte Carlo computer code MCNP [(2)] calculates the dose rates at the specified locations around the array of casks. The results of this calculation provide an example of how to demonstrate compliance with the relevant radiological requirements of 10 CFR 20 [(5)], 10 CFR 72[(4)], and 40 CFR 190[(6)] for a specific site. Each site must perform specific site calculations to account for the actual layout of the casks and fuel source. Two independent methods are utilized to evaluate the dose rate as a function of distance from the ISFSI. These are the "Blocking Factor" method and the "MCNP Array" method. These methods and their results are described below.

10.2.2.1 Blocking Factor Method – Methodology and Assumptions

The blocking factor method is a two step method to determine the dose rate as a function of distance from the ISFSI for an array of casks. The first step involves the calculation of the dose rate as a function of distance for a single cask. For the purpose of expanding the single cask results to be applicable for a multi-cask ISFSI, the total dose rate at any given location is a sum of dose rate contribution from two components – *direct* and *skyshine*. The direct dose rate component is basically due to the unobstructed (no shielding from other casks) sources. The skyshine component is due to the scattering of source particles around (and above) the cask. A conservative assumption in the blocking factor approach is that the dose rate contribution from skyshine is not blocked. That is, the blocking factors are applied only to the direct dose rate component. Two different MCNP calculations are performed to determine the total and direct dose rates. In the total dose rate calculation model, the single cask ISFSI geometry is modeled in its entirety. The results of these calculations are shown in Chapter 5. In the direct dose rate calculational model, the top boundary is typically set at 500 cm or any other short distance from the cask top. This ensures that the scattering and particle transport above the cask top is minimized or eliminated, thereby removing the skyshine component. The conservatism in the direct model depends on the distance of the top boundary from the cask top. A typical value for this distance is the inter-cask radial separation distance. The skyshine doses are calculated by subtracting the direct doses from the total doses. Two models are used to determine the dose rates that differ on the source term – a "gamma" model to calculate the gamma dose rates due to the fuel and hardware and a "neutron" model to determine the neutron dose rates from the active fuel. All the doses are determined using volumetric F4 detectors.

In the second step of the methodology, the blocking factors required to scale the single cask direct dose rates as a function of the cask location and distance are calculated. Subsequently, the total dose rates for the ISFSI are calculated as a cumulative sum of the direct and skyshine dose rate components for all the casks in the ISFSI. The QADS module of the SCALE4.4 [(3)] computer code system is utilized to determine the blocking factors.

The assumptions for the blocking factor methodology are summarized below.

- Because the cask height is not a factor for a radial point kernel calculation, the casks are 10 cm high in the QADS model of the array. The shielding calculations are performed using

iron cross sections with a Co60 source and air scattering. The iron cross sections are sufficient since the bulk of the cask shielding materials are steels. The gamma source in the cask is well approximated using the Co60 source.

- The dose rates due to capture gamma sources are not calculated since they are insignificant at large distances in comparison to primary gamma and neutron sources.
- The "ground shine" contribution is not calculated since soil is not modeled explicitly beyond the concrete pad of the ISFSI. The ground shine component is that portion of the dose rate that is due to reflection/scatter from the ground. The ground shine is significant only at the immediate vicinity of the cask (source) and is relatively insignificant at far distances.
- The choice of the F4 annular cylindrical detectors for the single cask MCNP model tallies inherently assumes that there is no effect due to the orientation of the casks on the dose rates especially at far distances.
- The selection of the top boundary for determining the direct doses in the blocking factor MCNP model is conservative and is expected to result in a higher skyshine component.

Blocking Factor Method - MCNP and QADS Results

The *blocking factor* MCNP model consists of a single TN 68 cask that is centered on a 14-foot concrete pad. The problem geometry is extended to include volumetric F4 detectors placed at 10m, 20m, 40m, 60m, 80m, 100m, 200m, 400m and 600m from the edge of the concrete pad. The detectors are modeled as annular cylinders with a thickness of 30cm and an axial height of 30cm. For calculating the total dose rates, the top boundary of the model is extended to 50000 cm which provides adequate room for scattering of particles in air. In the direct dose calculational model, the top boundary is fixed at 500 cm, which is about 6 ft from the top of the cask. This distance is also the approximate distance of separation between casks in the ISFSI. The choice of 500 cm as the top boundary is conservative since a realistic value would be about twice the distance thereby providing for a free air space above the cask that is equal to a single cask height. The implication of a conservative axial boundary is that any particle that crosses the boundary is lost and does not contribute to the direct dose tally. Therefore, this model is expected to result in an overestimation of the skyshine dose rates and an underestimation of the blocking or cask-shielded dose rates.

Table 10.2-1 shows the results of the MCNP farfield (total and direct) dose rate calculations for a single cask. The skyshine dose rate calculations are shown in Table 10.2-2. Skyshine dose rates are obtained by subtracting the direct dose rates from the total dose rates. In order to obtain a smooth fit of the data for use in the subsequent ISFSI dose evaluations, the gamma skyshine dose rates are adjusted conservatively. These adjusted dose rates are shown in column 3 of Table 10.2-2. A blank entry in these columns indicates that the calculated dose rates are utilized to determine the skyshine dose rates (no adjustments are made) as a function of distance. Only the dose rates at 10m and 40m distance are adjusted.

The MCNP results are utilized to determine mathematical equations that express the dose rate as a function of distance for both gamma and neutron sources. Due to the large distances involved, two equations are determined to represent these dose rates – short distance (0 – 80 m) and long

distance (80 – 600m). Figure 10.2-2 shows plots of the total dose rates as a function of distance. The plot at the top is based on short distance data and the one at the bottom is based on the long distance data. The mathematical function for gamma dose rates is shown at the top of each plot while that for neutron is shown at the bottom of each plot. The R^2 value (included in the plots) of these functions indicates that they represent a very good fit of the dose rates. Figure 10.2-3 shows plots of the direct dose rates as a function of distance and Figure 10.2-4 shows plots of the skyshine dose rates as a function of distance. The mathematical equations developed in the previous sections that fit the MCNP dose rate results as a function of distance for a single cask are summarized in Table 10.2-3.

Due to the layout of the ISFSI, three types of blocking factors are determined – longer side, shorter side and corner. The longer side of the cask array is that side that has 10 casks facing it and is shown in Figure 10.2-1. The dose points are located at 10m, 20m, 40m, 60m, 80m, 100m, 200m, 400m, and 600m from the edge of the ISFSI array between casks 5 and 6. The shorter side of the cask array is that side that has 2 casks facing it and is also shown in Figure 10.2-1. The dose points are located at 10m, 20m, 40m, 60m, 80m, 100m, 200m, 400m, and 600m from the edge of the ISFSI array between casks 1 and 11. The dose calculational methodology is similar to that outlined for the longer array of casks. The corner dose points are located 10m, 20m, 40m, 60m, 80m, 100m, 200m, 400m, and 600m from the corner of cask 1. Due to the symmetry of the ISFSI along the sides, only the blocking factors for casks 11 through 15 were determined along the longer side of the array. Along the shorter side of the array, the blocking factor for cask 15 in the longer direction (same as the blocking factor for cask 11 in the shorter direction) - is conservatively utilized as the blocking factor for casks 2 through 10. The unblocked casks for the longer side of array are casks 1 through 10 while the unblocked casks for the shorter side of array are casks 1 and 11. The unblocked casks for the corner dose points are casks 1 through 11.

The blocking factor calculation concept is pictorially represented in Figure 10.2-5 for Cask 13 at 10m distance (for longer side of the array). To calculate the blocking factor for cask 13, the casks 13, 14, 4 and 5 are represented in QADS with the location of Cask 13 at X=0, Y=0. Basically, all the casks that are likely to "block" cask 13 from the dose point A are modeled in QADS. Dose point A represents the actual location of the detector while the dose point B is a complementary position of the detector without any blocking. The ratio of the dose rates at dose point A and dose point B is what is called the "blocking factor".

As an example, at 10 m distance, the dose rate at dose point A is 0.514 and the dose rate at dose point B is 0.972. The blocking factor, at 10m for cask 13, is therefore 0.529 (0.514/0.972). All the results shown in this section are based on the same concept of calculating blocking factors with QADS. The blocking factors for the side and corner locations are shown in Table 10.2-4 and Table 10.2-5 respectively.

<u>Blocking Factor Method – Dose Rate as a Function of Distance</u>

Utilizing the mathematical equations and the blocking factors determined in the previous section for direct and skyshine dose rates as a function of distance, the dose rate contribution from each cask in the ISFSI and subsequently, the total dose rate as a function of distance for an array of casks can be determined. First, the direct dose rates and skyshine dose rates (for both gamma

and neutron) for the given distance is calculated using the equations determined in Table 10.2-3. Then, the direct dose rates are multiplied (or scaled) using the appropriate blocking factors determined in Table 10.2-4 and Table 10.2-5. Finally, the dose rate contributions are calculated as a summation of the skyshine and the blocked direct dose rates for each blocked cask. These results are shown in Table 10.2-6.

10.2.2.2 MCNP Array Method – Methodology, Model and Assumptions

The ISFSI layout as illustrated in Figure 10.2-1 is explicitly modeled in MCNP using advanced MCNP geometry. The gamma and neutron dose rates are determined as a function of distance from the ISFSI. All the doses are determined using F5 point detectors. The *ISFSI array* MCNP model consists of a 2x10 ISFSI containing TN 68 HB casks. The cask array is modeled using advanced MCNP geometry to represent the ISFSI as shown in Figure 3.2-1. The concrete pad at the bottom is modeled to span the extent of the array (X±2140 cm, Y±428cm). Three sets of point detector tallies are utilized to determine the dose rates. One set is utilized to determine the dose rate as a function of distance from the longer side of the array (between casks 5 and 6), the other is utilized to determine the dose rates from the shorter side of the array (between casks 1 and 11) and the third is utilized to determine the dose rates from the corner of the array (corner of cask 1). These point F5 detectors placed at 10m, 20m, 40m, 60m, 80m, 100m, 200m, 400m and 600m from the edge of the concrete pad at each direction (long, short and corner).

The assumptions for the MCNP methodology are summarized below.

- The doses due to capture gamma sources are not calculated since they are insignificant at large distances in comparison to primary gamma and neutron sources.
- The "ground shine" contribution is not calculated since soil is not modeled explicitly beyond the concrete pad of the ISFSI. The ground shine component is that portion of the dose rate that is due to reflection/scatter from the ground. The ground shine is significant only at the immediate vicinity of the cask (source) and is relatively insignificant at far distances.
- The location of the F5 detectors for the ISFSI array MCNP model inherently assume that there is no effect on the orientation of the casks on the dose rates especially at far distances.
- The "universe" is a cylinder surrounding the ISFSI. To account for skyshine radius of this sphere (r=150,000 cm) is more than 10 mean free paths for neutrons and 50 mean free paths for gammas greater than that of the outermost surface, thus ensuring that the model is of a sufficient size to include all interactions, including skyshine, affecting the dose rate at the detectors.

MCNP Array Method – Dose Rate as a Function of Distance

The MCNP results for each detector provides the dose rate as a function of distance at all the locations (sides and corner) around the ISFSI. These results are shown in Table 10.2-7. Some of the MCNP tally results appear to have very large errors associated with them. An inspection of the MCNP output indicates that there is expected to be no change in the tally value at convergence. However, for conservatism, the tallies with errors greater than 10% but less than 20% (neutron and gamma 600m tally for longer side, neutron 400m and gamma 600m tally for

corner) are scaled by a factor of 1.1 and the tallies for errors greater than 20% (neutron 400m and gamma 600m tally for shorter side) are scaled by a factor of 1.2.

10.2.2.3 ISFSI Annual Doses

The ISFSI annual doses (mrem) as a function of distance for both the methods are shown in Table 10.2-8. These doses are obtained by multiplying the dose rates (mrem/hour) with 8760 (total number of hours per year assuming full occupancy). These results are also shown pictorially in Figure 10.2-6 and Figure 10.2-7. The results indicate that the blocking factor methodology results in conservative dose rates. The ratio of the dose rates (blocking factor to MCNP array) is also shown in Table 10.2-8. These results also show that the longer side of the array and corner of array results for the blocking factor methodology are in better agreement to the MCNP array results than the shorter side of the array results. The agreement of blocking factor results with those of the MCNP array results is directly related to the number of blocked casks. The more the number of blocked casks, the higher the conservatism in the skyshine dose and therefore, the larger the ratio of the blocked dose rates to the MCNP array dose rates.

These results demonstrate that both methods result in similar dose rate predictions. The blocking factor methodology can be utilized to quickly determine the site doses even for complicated ISFSI layouts.

The preceding analyses and results are intended to provide high estimates of dose rates for generic ISFSI layouts. The written evaluations performed by a licensee for an actual ISFSI must consider the type and number of casks, layout, characteristics of the irradiated fuel to be stored, site characteristics (e.g., berms, distance to the controlled area boundary, etc.), and reactor operations at the site in order to demonstrate compliance with 10 CFR 72.104.

10.3 Estimated Onsite Collective Dose Assessment

Cask Loading Operations

Table 10.3-1 shows the estimated design basis occupational exposures to ISFSI personnel during the loading, transport, and emplacement of the storage casks (time and manpower may vary depending on individual utility practices). The task times, number of personnel required and the average distance from the cask are listed in this table.

This estimate of operational doses is based on design basis 8x8 fuel, DBF-68 as defined in Chapter 5. It assumes that the one inch thick auxiliary shield ring is installed, but there is no temporary shielding used. Lead bean bags and temporary plastic neutron shielding can be used to maintain doses ALARA.

Operations with the TN-68 have yielded much lower doses, as shown in Table 10.3-3, which shows cumulative dose measurements for loads of about 16 kW per cask. For the design basis load of 30 kW, the operational doses would be a factor of 2 to 4 higher.

The average distance for a given operation takes into account the fact that the operator may be momentarily in contact with the cask, but this time will be limited. For example, during bolting, the placement of the bolts in the holes will bring the operator in contact with the cask. While torquing the operator will be further away due to the typical length of a torque wrench handle. Similarly, for draining, vacuum drying, and leak testing, the attachment of fittings will take place closer to the cask than the operation of the pump and vacuum drying system. For decontamination, although operators will be close to the cask to take swipes, other parts of the operation will be done by hosing the cask down from further away.

For this reason, 0.5 or 1.0 meter is an appropriate average distance for these hands-on operations.

The operator's hands may be in a high dose rate location momentarily, for example when connecting couplings or vacuum fitting at the ports. This does not translate into a whole-body dose, and therefore, these localized streaming effects are not considered here.

For the operations near the lid, typically most of the operation will take place around the perimeter (corner) and a smaller portion will take place directly over the lid. A 33/67 weighted average of axial centerline and above neutron shield radial dose rates is used for these operations as described below.

Dose rates used for the operations dose estimate

All of the following dose rates are in mrem/hr. See also Figure 10.3–1.

Water/lid: Dose rates at the cask top while the cask is still filled with water are low due to the water shielding; they are estimated at

0.5 meter	11 γ / 12 n
2 meter	3 γ / 4 n

Lid/Corner: (prior to placement of top neutron shield) 33% axial dose rates at the cask lid centerline and 67% radial dose rate above the neutron shield:

0.5 meter:	325 γ / 45 n
1 meter	178 γ / 18 n

Top/Corner (after installation of top neutron shield): the radial dose rates above the neutron shield are taken from Tables 5.4-2 and 5.4-3 and interpolated as necessary.

0.5 meter:	232 γ / 35 n
1 meter	87 γ / 11 n

Radial (midplane dose rates from Table 5.4-2, interpolated and extrapolated as necessary)

0.5 meter	85 γ / 17 n
1 meter:	57 γ / 11 n
2 meter	34 γ / 6.5 n
3 meter	23 γ / 4.2 n

Maintenance Operations

Table 10.3-2 shows the estimated annual person-rem for surveillance and maintenance activities. These estimates take no credit for reduced does rates due to decay time at the ISFSI. The background dose rate at the ISFSI is estimated at 15γ /2.8n mrem/hr based on a distance of more than 4 meters from the nearest cask, except as noted. Dose rates from the nearest cask are based upon the radial midplane dose rates for 30 kW cask loads except for repairs under the protective cover, which consider dose rates at the top of the cask.

For operability tests and calibration, and for unanticipated instrument repair, the worker was assumed to be located at the plumbing manifold located on the cask exterior about 4 feet from the ground, an average of 1 meter from the cask. Repressurization of the overpressure system may be done at the same time as calibration with little or no additional exposure.

For paint touch up, an average distance is 0.5 meter.

For major repairs to the overpressure system that would require removal of the weather protective cover, the 0.5 meter radial dose rate from the area above the radial neutron shield is used (top/corner dose rate).

10.4 References

1. U.S. Nuclear Regulatory Commission, Regulatory Guide 8.8, Information Relevant to Ensuring that Occupational Exposures at Nuclear Power Stations will be As Low As Is Reasonably Achievable, Revision 3, June 1978.

2. MCNP4C2, Monte Carlo N-Particle Transport Code System, Oak Ridge National Laboratory, CCC-701, RSICC Computer Code Collection, June 2001

3. "SCALE, A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation", NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), Vol. I-III, September 1998.

4. Title 10 Code of Federal Regulations Part 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste

5. Title 10 Code of Federal Regulations Part 20, Standards for Protection Against Radiation

6. Title 40 Code of Federal Regulations Part 190, Environmental Radiation Protection Standards for Nuclear Power Operations

TABLE 10.2-1

TOTAL AND DIRECT DOSE RATES AS A FUNCTION OF DISTANCE FOR A SINGLE TN-68 CASK

Distance (meters)	Gamma (mrem/hour)	Error	Neutron (mrem/hour)	Error
Total Dose Rate Results				
10	3.67E+00	0.0199	7.14E-01	0.0198
20	1.21E+00	0.0272	2.23E-01	0.0240
40	3.07E-01	0.0330	5.27E-02	0.0279
60	1.34E-01	0.0365	2.36E-02	0.0307
80	7.124E-02	0.0398	1.16E-02	0.0302
100	4.06E-02	0.0444	6.74E-03	0.0323
200	5.30E-03	0.0513	8.88E-04	0.0362
400	4.42E-04	0.0547	5.62E-05	0.0313
600	6.50E-05	0.0752	6.79E-06	0.0460
Direct Dose Rate Results				
10	3.65E+00	0.0197	6.61E-01	0.0187
20	1.14E+00	0.0259	1.95E-01	0.0242
40	2.91E-01	0.0340	3.98E-02	0.0336
60	1.13E-01	0.0409	1.50E-02	0.0427
80	5.73E-02	0.0454	6.39E-03	0.0503
100	3.18E-02	0.0505	3.11E-03	0.0583
200	3.45E-03	0.0780	2.84E-04	0.1080
400	1.73E-04	0.1278	8.12E-06	0.1834
600	2.41E-05	0.1736	7.11E-07	0.2947

TABLE 10.2-2

SKYSHINE DOSE RATES AS A FUNCTION OF DISTANCE FOR A SINGLE TN-68 CASK

Distance (meters)	Gamma (mrem/hour)	Adjusted Gamma (mrem/hour)	Neutron (mrem/hour)
10	2.12E-02	1.20000E-01	5.28E-02
20	7.33E-02		2.81E-02
40	1.54E-02	3.08809E-02	1.29E-02
60	2.12E-02		8.57E-03
80	1.38E-02		5.21E-03
100	8.79E-03		3.63E-03
200	1.85E-03		6.04E-04
400	2.69E-04		4.81E-05
600	4.09E-05		6.08E-06

TABLE 10.2-3

MATHEMATICAL FUNCTIONS FOR DOSE RATES AS A FUNCTION OF DISTANCE FOR A SINGLE TN-68 CASK

Description	Mathematical Function
X=Distance from Edge of PAD, X ≤ 80 m	
Total Dose Rate, Gamma	$323.62*(X)^{(-1.9035)}$
Total Dose Rate, Neutron	$74.803*(X)^{(-1.9789)}$
Direct Dose Rate, Gamma	$405.71*(X)^{(-1.9973)}$
Direct Dose Rate, Neutron	$129.23*(X)^{(-2.2236)}$
Skyshine Dose Rate, Gamma	$1.4694*(X)^{(-1.0462)}$
Skyshine Dose Rate, Neutron	$0.6961*(X)^{(-1.0921)}$
X=Distance from Edge of PAD, X ≥ 80 m	
Total Dose Rate, Gamma	$2.8474*10^5*(X)^{(-3.4209)}$
Total Dose Rate, Neutron	$1.2938*10^5*(X)^{(-3.6374)}$
Direct Dose Rate, Gamma	$1.5472*10^6*(X)^{(-3.8439)}$
Direct Dose Rate, Neutron	$2.8503*10^6*(X)^{(-4.4683)}$
Skyshine Dose Rate, Gamma	0.0247*EXP((X)*(-0.011))
Skyshine Dose Rate, Neutron	0.0116*EXP((X)*(-0.013))

TABLE 10.2-4

BLOCKING FACTORS FOR USE IN THE SIDE (SHORTER AND LONGER) DOSE RATE CALCULATIONS

Distance	Cask 15	Cask 14	Cask 13	Cask 12	Cask 11
10 m	0.100	0.705	0.529	0.529	0.529
20 m	0.015	0.448	0.706	0.623	0.331
40 m	0.013	0.245	0.292	0.499	0.688
60 m	0.012	0.106	0.262	0.291	0.483
80 m	0.011	0.099	0.218	0.269	0.291
100 m	0.001	0.088	0.110	0.254	0.273
200 m	0.001	0.014	0.087	0.101	0.108
400 m	0.001	0.012	0.014	0.015	0.087
600 m	0.001	0.001	0.012	0.013	0.015

TABLE 10.2-5

BLOCKING FACTORS FOR USE IN THE CORNER DOSE RATE CALCULATIONS

Distance	Cask 12	Cask 13	Cask 14	Cask 15	Casks 16 through 20
10 m	0.001	0.152	0.001	0.001	0.001
20 m	0.001	0.158	0.001	0.001	0.001
40 m	0.001	0.047	0.002	0.001	0.001
60 m	0.001	0.045	0.157	0.137	0.001
80 m	0.001	0.044	0.151	0.107	0.143
100 m	0.001	0.041	0.049	0.157	0.017
200 m	0.001	0.001	0.044	0.047	0.137
400 m	0.001	0.001	0.001	0.042	0.045
600 m	0.001	0.001	0.001	0.001	0.042

TABLE 10.2-6

DOSE RATE AS A FUNCTION OF DISTANCE FOR THE ISFSI (BLOCKING FACTOR METHOD)

Distance (m)	**Gamma (mrem/hr)**	**Neutron (mrem/hr)**	**Total (mrem/hr)**	**Gamma (mrem/hr)**	**Neutron (mrem/hr)**	**Total (mrem/hr)**
	Longer Side of Array			Shorter Side of Array		
10	3.25E+01	6.24E+00	3.88E+01	1.14E+01	2.41E+00	1.38E+01
20	1.19E+01	2.20E+00	1.41E+01	3.10E+00	7.44E-01	3.85E+00
40	3.66E+00	6.72E-01	4.33E+00	1.11E+00	2.99E-01	1.41E+00
60	1.70E+00	3.20E-01	2.02E+00	6.28E-01	1.81E-01	8.08E-01
80	1.03E+00	1.96E-01	1.23E+00	4.25E-01	1.27E-01	5.52E-01
100	5.18E-01	1.01E-01	6.19E-01	2.26E-01	6.87E-02	2.95E-01
200	6.54E-02	1.37E-02	7.91E-02	5.62E-02	1.63E-02	7.25E-02
400	6.50E-03	1.05E-03	7.54E-03	6.13E-03	1.23E-03	7.36E-03
600	1.22E-03	1.46E-04	1.36E-03	7.81E-04	1.05E-04	8.86E-04

Distance (m)	**Gamma (mrem/hr)**	**Neutron (mrem/hr)**	**Total (mrem/hr)**
	Corner of Array		
10	1.41E+01	2.75E+00	1.69E+01
20	5.71E+00	1.09E+00	6.80E+00
40	2.15E+00	4.11E-01	2.56E+00
60	1.19E+00	2.27E-01	1.41E+00
80	7.15E-01	1.37E-01	8.52E-01
100	3.74E-01	7.34E-02	4.47E-01
200	5.56E-02	1.11E-02	6.67E-02
400	5.81E-03	8.94E-04	6.70E-03
600	1.17E-03	1.38E-04	1.31E-03

TABLE 10.2-7

DOSE RATE AS A FUNCTION OF DISTANCE FOR THE ISFSI (MCNP ARRAY METHOD)

Distance (meters)	Gamma (mrem/hour)	Error	Neutron (mrem/hour)	Error	Total (mrem/hour)
Longer Side Dose Rate Results					
10	3.08E+01	0.0391	5.34E+00	0.0154	3.61E+01
20	1.31E+01	0.0350	2.31E+00	0.0376	1.54E+01
40	3.88E+00	0.0325	6.84E-01	0.0160	4.56E+00
60	1.57E+00	0.0434	2.89E-01	0.0173	1.86E+00
80	8.19E-01	0.0599	1.57E-01	0.0278	9.76E-01
100	4.83E-01	0.0436	8.69E-02	0.0240	5.70E-01
200	7.68E-02	0.0567	1.20E-02	0.0392	8.88E-02
400	5.46E-03	0.0995	7.45E-04	0.0667	6.20E-03
600	8.30E-04	0.1383	1.22E-04	0.1967	9.52E-04

Shorter Side Dose Rate Results					
10	7.46E+00	0.0583	2.13E+00	0.0243	9.59E+00
20	2.50E+00	0.0523	8.15E-01	0.0271	3.32E+00
40	7.68E-01	0.0461	2.89E-01	0.0918	1.06E+00
60	3.86E-01	0.0501	1.29E-01	0.0229	5.15E-01
80	2.28E-01	0.0484	7.22E-02	0.0200	3.00E-01
100	1.44E-01	0.0455	4.78E-02	0.0358	1.92E-01
200	2.53E-02	0.0466	7.48E-03	0.0333	3.28E-02
400	2.10E-03	0.0796	9.89E-04	0.3413	3.09E-03
600	5.26E-04	0.2952	5.51E-05	0.0926	5.82E-04

Corner Dose Rate Results					
10	1.17E+01	0.0747	2.40E+00	0.0578	1.41E+01
20	5.42E+00	0.0662	1.03E+00	0.0285	6.45E+00
40	2.02E+00	0.0597	3.91E-01	0.0292	2.41E+00
60	9.87E-01	0.0534	1.91E-01	0.0326	1.18E+00
80	5.61E-01	0.0495	1.10E-01	0.0459	6.71E-01
100	3.54E-01	0.0532	6.51E-02	0.0371	4.19E-01
200	5.45E-02	0.0550	9.59E-03	0.0580	6.41E-02
400	4.79E-03	0.1023	7.06E-04	0.1296	5.49E-03
600	7.73E-04	0.1791	6.17E-05	0.0862	8.34E-04

TABLE 10.2-8

ISFSI ANNUAL DOSES AS A FUNCTION OF DISTANCE FOR 20 TN-68 CASKS

Distance (m)	Blocking Total (mrem)	Arrays Total (mrem)	Ratio (Blocking/ Arrays)	Blocking Total (mrem)	Arrays Total (mrem)	Ratio (Blocking/ Arrays)
	Longer Side of Array			Shorter Side of Array		
10	3.39E+05	3.16E+05	1.073	1.21E+05	8.40E+04	1.436
20	1.23E+05	1.35E+05	0.913	3.37E+04	2.91E+04	1.159
40	3.79E+04	4.00E+04	0.949	1.23E+04	9.25E+03	1.331
60	1.77E+04	1.63E+04	1.088	7.08E+03	4.51E+03	1.570
80	1.07E+04	8.55E+03	1.256	4.83E+03	2.63E+03	1.839
100	5.43E+03	4.99E+03	1.086	2.58E+03	1.68E+03	1.539
200	6.93E+02	7.78E+02	0.891	6.35E+02	2.87E+02	2.211
400	6.61E+01	5.43E+01	1.216	6.45E+01	2.71E+01	2.384
600	1.19E+01	8.34E+00	1.430	7.76E+00	5.09E+00	1.524

Distance (m)	Blocking Total (mrem)	Arrays Total (mrem)	Ratio (Blocking/ Arrays)
	Corner of Array		
10	1.48E+05	1.24E+05	1.192
20	5.96E+04	5.65E+04	1.054
40	2.24E+04	2.11E+04	1.060
60	1.24E+04	1.03E+04	1.199
80	7.46E+03	5.88E+03	1.269
100	3.92E+03	3.67E+03	1.068
200	5.84E+02	5.61E+02	1.041
400	5.87E+01	4.81E+01	1.221
600	1.14E+01	7.31E+00	1.566

TABLE 10.3-1
OCCUPATIONAL EXPOSURES FOR CASK LOADING, TRANSPORT, AND EMPLACEMENT (ONE TIME EXPOSURE, 30 kW CONTENTS)

						GAMMA		NEUTRON	
		No of Persons	Time (hr)	Avg Dist (m)	location	Dose rate mrem/hr	person-rem	Dose rate mrem/hr	person-rem
A. Cask Receipt									
1-12	Unloading, inspection, etc.	NO EXPOSURE OTHER THAN BACKGROUND							
B. Cask Loading Pool									
1	Lower cask into cask loading pool	NO EXPOSURE OTHER THAN BACKGROUND (POOL)							
2	Load	NO EXPOSURE OTHER THAN BACKGROUND (POOL)							
3	Verify	NO EXPOSURE OTHER THAN BACKGROUND (POOL)							
4	Lower lid	NO EXPOSURE OTHER THAN BACKGROUND (POOL)							
5	Lift cask and install some of the lid bolts hand tight	1	0.25	0.5	water/lid	11	0.0026	12	0.0030
		1	0.5	2	water/lid	3	0.0015	4.0	0.0020
6	Drain (pump) water	1	0.5	0.5	water/lid	11	0.0053	12	0.0060
		1	1	2	water/lid	3	0.0030	4.0	0.0040
7	Disconnect drain line	1	0.25	0.5	lid/corner	325	0.0812	45	0.0114
8	Move to decontamination area	2	1	2	radial	34	0.0688	6.5	0.0130
C. Decontamination Area									
1	Decontaminate	2	1	1	radial	57	0.1134	11	0.0220
		1	0.5	1	lid/corner	178	0.0888	18	0.0092
2	Install remaining lid bolts and torque	2	1	0.5	lid/corner	325	0.6495	45	0.0910
3	Remove plug from neutron shield vent, install pressure relief valve.	1	0.25	0.5	lid/corner	325	0.0812	45	0.0114
4	Connect the Vacuum Drying System	1	0.25	0.5	lid/corner	325	0.0812	45	0.0114
		1	0.5	2	radial	34	0.0172	6.5	0.0032
5	Continue vacuum drying	1	1	1	radial	57	0.0567	11	0.0110
6	Backfill cask with helium and pressurize	1	0.25	0.5	lid/corner	325	0.0812	45	0.0114
		2	1	2	radial	34	0.0688	6.5	0.0130
7	Helium leak test all lid and port cover seals	1	1	0.5	lid/corner	325	0.3248	45	0.0455
		2	2	2	radial	34	0.1376	6.5	0.0260
8	Install top neutron shield.	2	0.25	0.5	top/corner	232	0.1160	35	0.0175
9	Install overpressure system tank	2	0.5	0.5	top/corner	232	0.2320	35	0.0350
10	Leak test OP system	2	0.5	1	top/corner	87	0.0872	11	0.0114
11	Pressurize OP system	1	0.5	1	top/corner	87	0.0436	11	0.0057
12	Install protective cover	2	1	0.5	top/corner	232	0.4640	35	0.0700
13	Check surface dose rate and contamination levels	2	0.5	1	radial	57	0.0567	11	0.0110
14	Install exterior tubing, leak test	1	0.5	0.5	top/corner	232	0.1160	35	0.0175
		1	1	1	radial	57	0.0567	11	0.0110
15	Backfill exterior tubing	1	0.5	1	radial	57	0.0284	11	0.0055
16	Load cask on transporter	2	1	2	radial	34	0.0688	6.5	0.0130
17	Move cask to storage area	2	3	3	radial	23	0.1368	4.2	0.0252
D. Storage Area									
1	Lower cask onto storage pad	2	0.5	2	radial	34	0.0344	6.5	0.0065
2	Disconnect cask transporter	2	0.5	2	radial	34	0.0344	6.5	0.0065
3	Connect over pressure system to monitoring panel	2	1	1	radial	57	0.1134	11	0.0220
4	Check OP system function.	1	0.5	1	radial	57	0.0284	11	0.0055
						Total	3.48 γ		0.56
						n+gam			4.04

TABLE 10.3-2

ISFSI MAINTENANCE OPERATIONS
ANNUAL EXPOSURES FOR 30 kW CONTENTS

GAMMA

Task	Time Req'd (hr)	No of Person	Dist (m)	Dose Rate (mrem/hr)	Backgrnd (mrem/hr)	Operation Dose (rem)	Operation Frequency (/year)	Annual Dose (rem)
Visual Surveillance of Casks	0.25	1	2	34	15	0.0124	12	0.148
Instrumentation								
a. Operability & Calibration	2	2	1	57	15	0.287	1	0.287
b. Unanticipated Repairs	2	2	1	57	15	0.287	0.25	0.072
Surface Defect Repair	1	2	0.5	85	15	0.199	1	0.199
Repair under Protective Cover	8	2	0.5	232	15	3.952	0.05	0.198

NEUTRON AND TOTAL

Task	Time Req'd (hr)	No of Person	Dist (m)	Dose Rate (mrem/hr)	Backgrnd (mrem/hr)	Operation Dose (rem)	Operation Frequency (/year)	Annual Dose (rem)	Total gamma + n (rem)
Visual Surveillance of Casks	0.25	1	2	6.5	2.8	0.0023	12	0.028	0.176
Instrumentation									
a. Operability & Calibration	2	2	1	11	2.8	0.0552	1	0.055	0.342
b. Unanticipated Repairs	2	2	1	11	2.8	0.0552	0.25	0.014	0.086
Surface Defect Repair	1	2	0.5	17	2.8	0.0393	1	0.039	0.239
Repair under Protective Cover	8	2	0.5	35	2.8	0.6048	0.05	0.030	0.228

1. All dose rates assume that the TN-68 cask contains design basis fuel. No reduction of dose rate is assumed for decay time.

2. Doses are on a per cask basis.

TABLE 10.3-3

MEASURED OPERATIONAL DOSES FROM TN-68 CASK LOADING

Year	Cask Number	Cask Heat Load kW	Man-Rem	Man-Hours
2000	TN-68-01	17.2	0.375	1744
2000	TN-68-02	17.1	0.284	1430
2000	TN-68-03	17.1	0.279	1272
2000	TN-68-04	17.1	0.168	1267
2001	TN-68-05	17.1	0.357	1068
2001	TN-68-06	17.1	0.358	937
2001	TN-68-07	17.2	0.371	1020
2001	TN-68-08	17.2	0.345	896
2001	TN-68-09	17.3	0.257	787
2002	TN-68-10	17.2	0.238	926
2002	TN-68-11	16.8	0.189	666
2002	TN-68-12	16.6	0.198	680
2002	TN-68-13	16.6	0.184	658
2002	TN-68-14	16.7	0.216	680
2002	TN-68-15	16.8	0.254	726
2003	TN-68-16	15.7	0.336	1104
2003	TN-68-17	16.5	0.206	559
2003	TN-68-18	16.5	0.254	724
2003	TN-68-19	16.6	0.198	589
2003	TN-68-20	16.6	0.235	592
2004	TN-68-21	16.8	0.226	710
2004	TN-68-22	17.0	0.208	604
2004	TN-68-23	16.9	0.210	575
2004	TN-68-24	17.0	0.183	560

Data provided by Peach Bottom Atomic Power Station

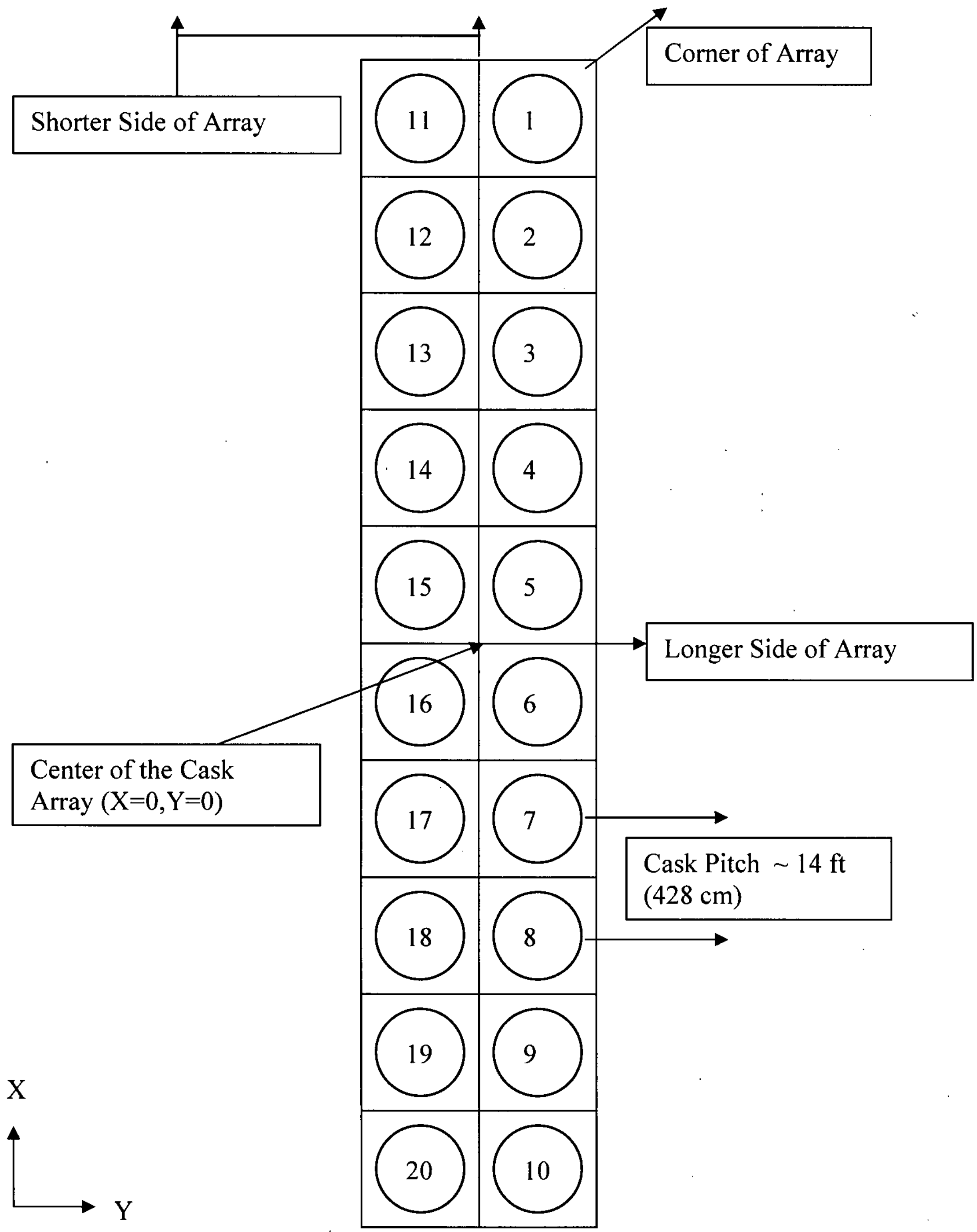
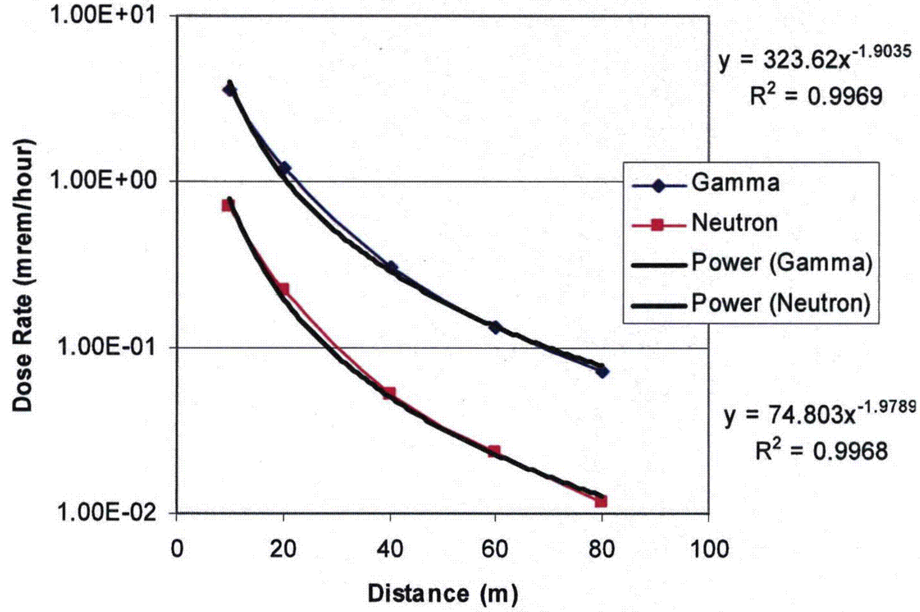


FIGURE 10.2-1

ISFSI LAYOUT WITH A 2X10 ARRAY OF TN-68 CASKS



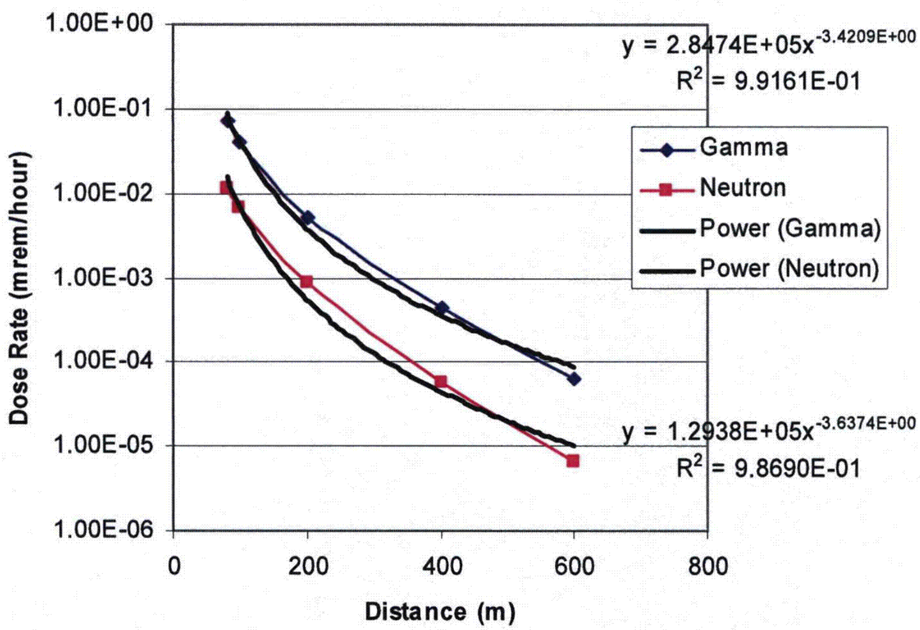
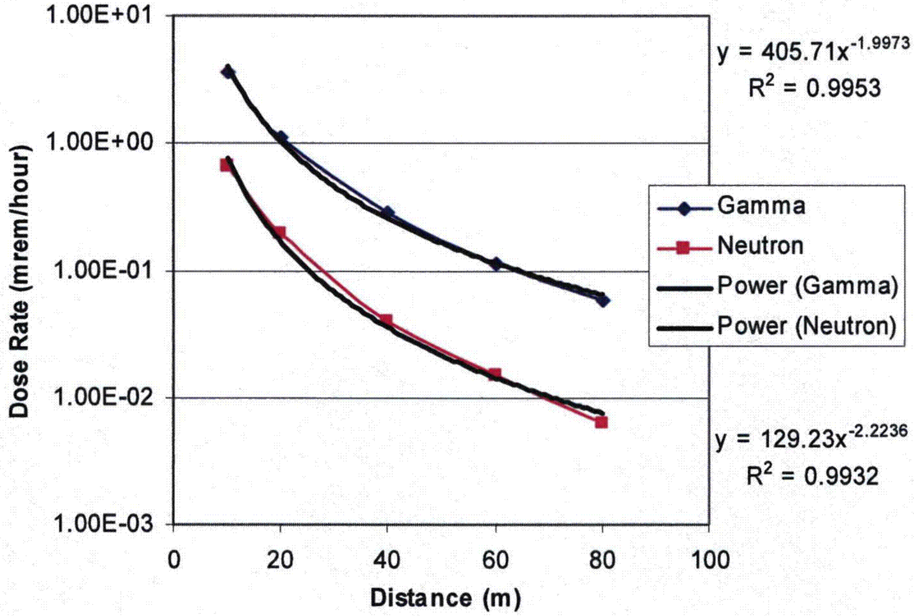


FIGURE 10.2-2

TOTAL DOSE RATE AS A FUNCTION OF DISTANCE FOR A SINGLE TN-68 CASK



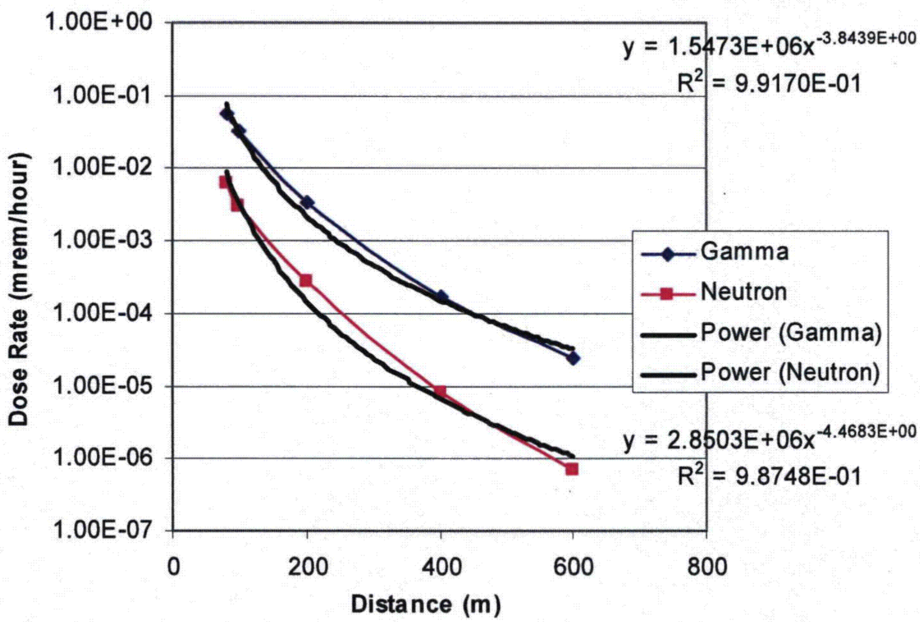
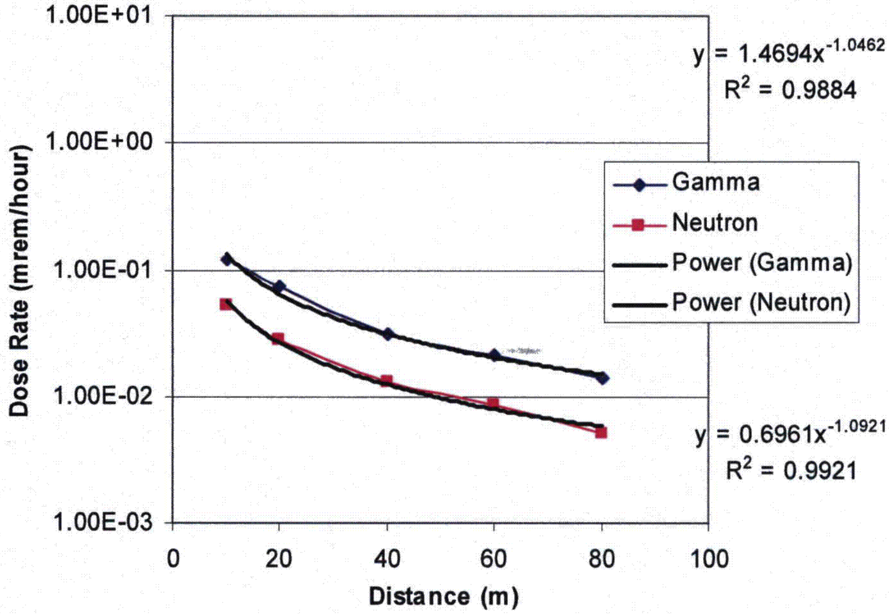


FIGURE 10.2-3

DIRECT DOSE RATE AS A FUNCTION OF DISTANCE FOR A SINGLE TN-68 CASK



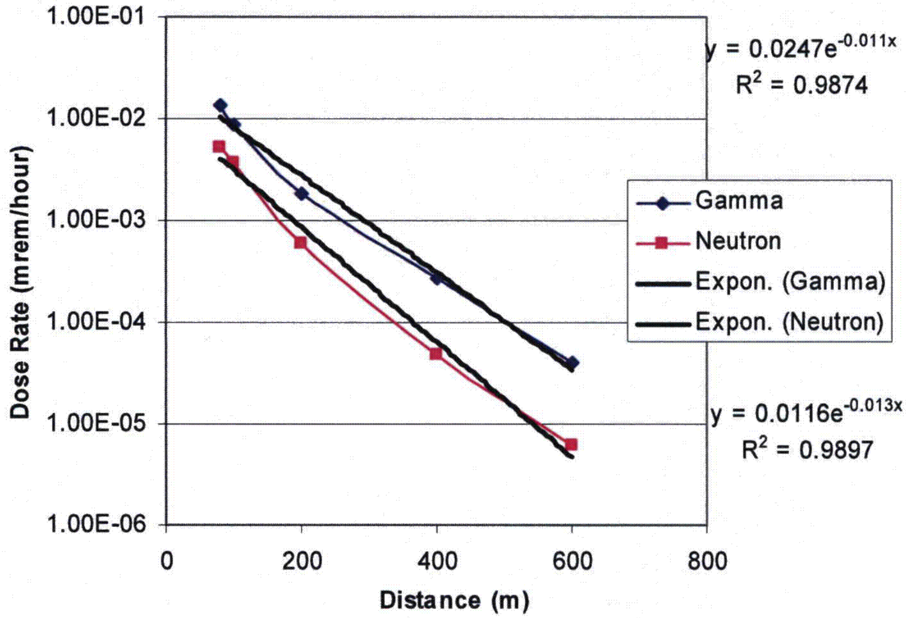


FIGURE 10.2-4

SKYSHINE DOSE RATE AS A FUNCTION OF DISTANCE FOR A SINGLE TN-68 CASK

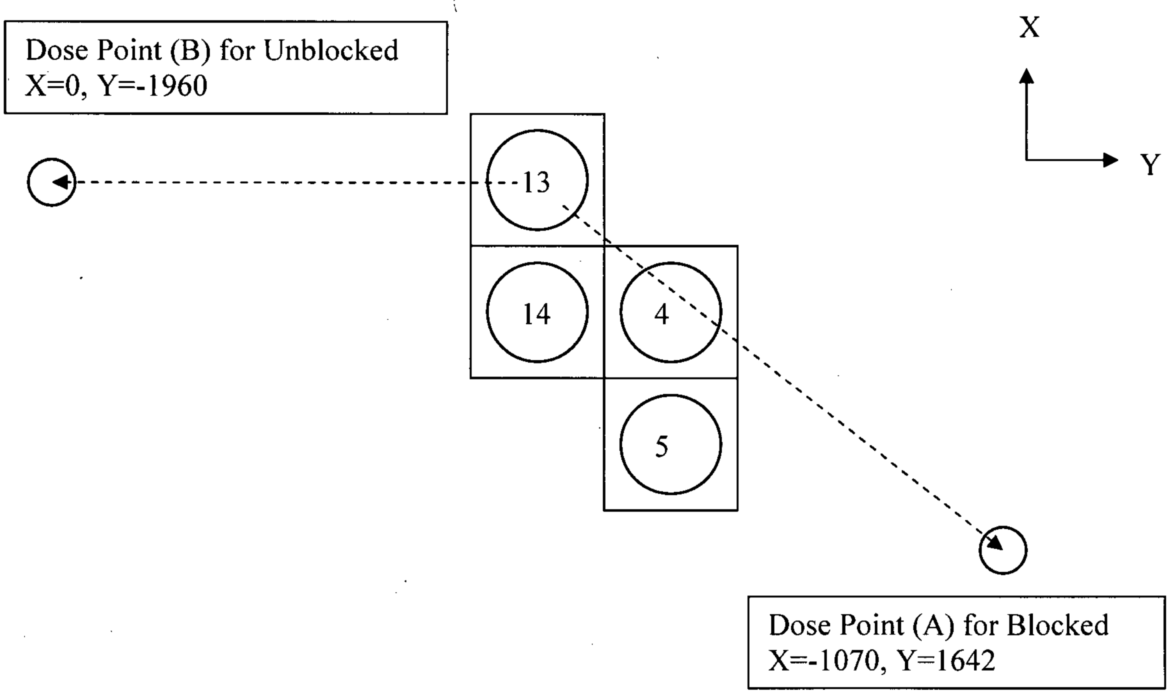
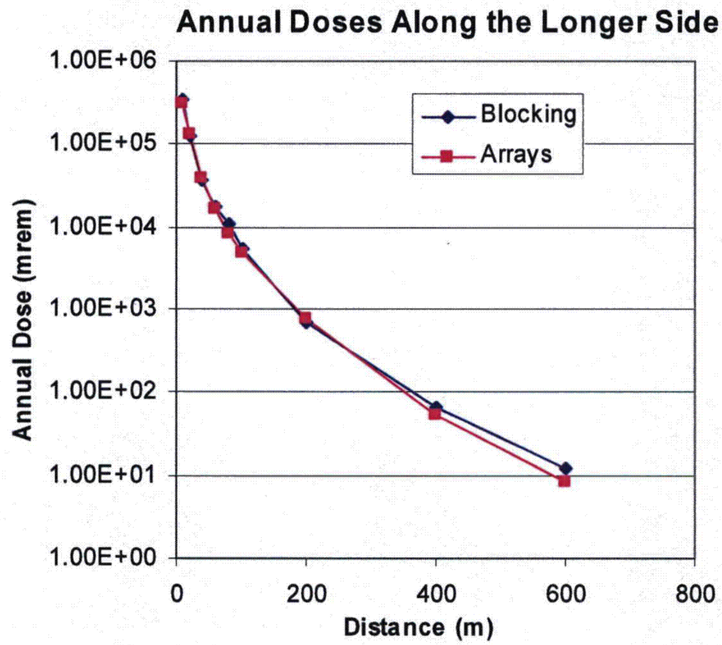


FIGURE 10.2-5

BLOCKING FACTOR CALCULATION ILLUSTRATION WITH QADS



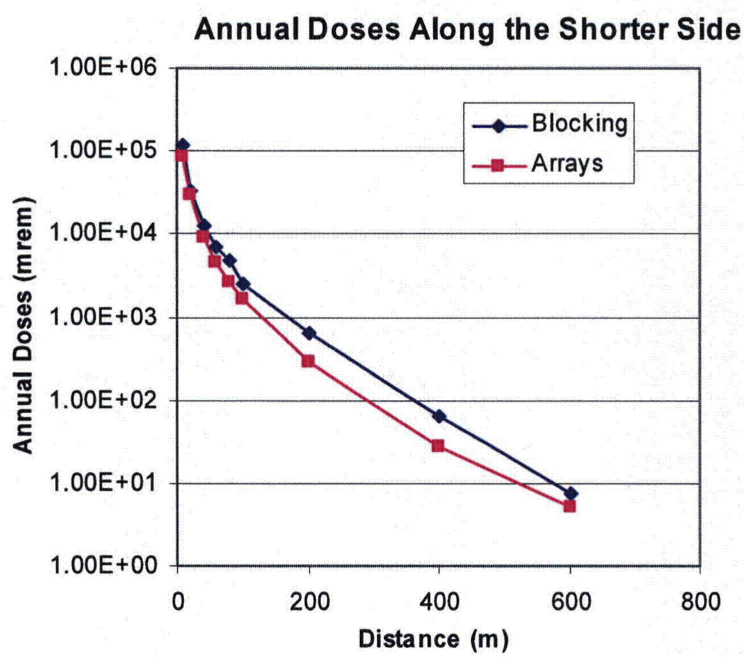


FIGURE 10.2-6

ISFSI SIDE ANNUAL DOSES AS A FUNCTION OF DISTANCE – 20 TN-68 CASKS

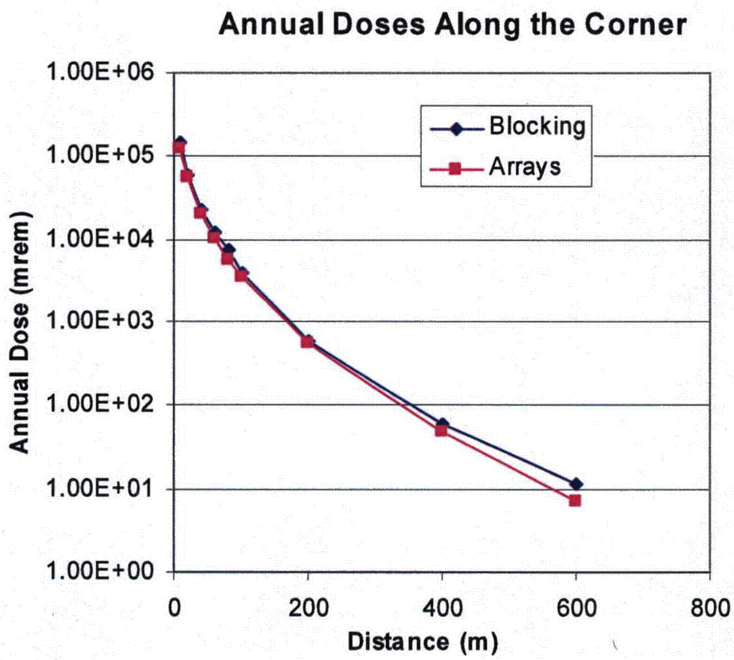
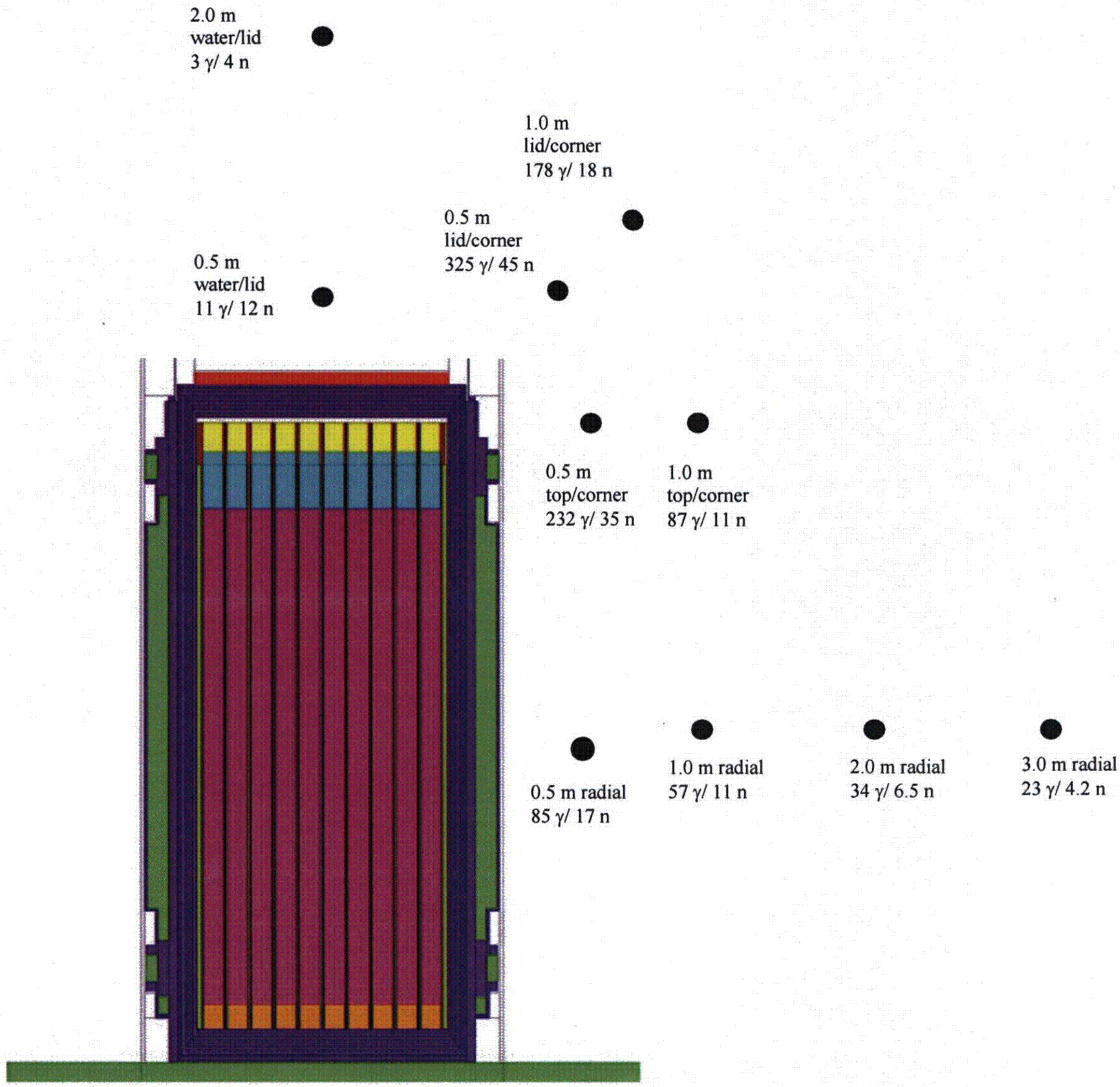


FIGURE 10.2-7

ISFSI CORNER ANNUAL DOSES AS A FUNCTION OF DISTANCE – 20 TN-68 CASKS



Notes:
1. Dose rates in mrem/hr
2. Water/lid before installation of top neutron shield, cask filled with water
3. Lid/corner before installation of top neutron shield, cask drained
4. Top/corner and radial in final cask configuration

FIGURE 10.3-1
DOSE RATES AND LOCATIONS
FOR TN-68 OCCUPATIONAL EXPOSURE ASSESSMENTS

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CHAPTER 11

ACCIDENT ANALYSES

This Chapter describes the postulated off-normal and accident events which could occur during storage of the TN-68 cask at an ISFSI. Detailed analysis of the events are provided in other SAR chapters and referenced herein.

11.1 Off-Normal Operations

Off-normal operations are design events of the second type (Design Event II) as defined in ANSI/ANS 57.9[1]. Design Event II conditions consist of that set of events that, although not occurring regularly, can be expected to occur with moderate frequency or on the order of once during a calendar year of ISFSI operation.

Two off-normal conditions have been considered with regard to the TN-68 cask, a loss of electric power and cask seal leakage.

11.1.1 Loss of Electric Power

A total loss of electric power to the ISFSI is postulated. The failure could be either an open or a short to ground circuit, or any other mechanism capable of producing an interruption of power.

11.1.1.1 Postulated Cause of the Event

A loss of power to the ISFSI may occur as a result of natural phenomena, such as lightning or extreme wind, or as a result of undefined disturbances in the nonsafety-related portion of the electric power system.

If electric power is lost, the following systems could be de-energized and rendered nonfunctional:

- Area lighting
- Cask pressure monitoring instrumentation

11.1.1.2 Detection of Event

A loss of power at the ISFSI site would be detected during periodic surveillance by noting that area lighting is not operational.

11.1.1.3 Analysis of Effects and Consequences

This event has no safety or radiological consequences because a loss of power will not affect the integrity of the storage casks, jeopardize the safe storage of the fuel, or result in radiological releases. None of the systems whose failure could be caused by this event are necessary for the accomplishment of the safety function of the cask. The lighting system functions merely for convenience and visual monitoring, and the instrumentation monitors the long-term performance of the storage casks with respect to the cask seals. None of these parameters are expected to change rapidly and their status is not dependent upon electric power.

A loss of power has no effect on the subcritical condition of the cask, cask confinement or retrievability of the fuel.

11.1.1.4 Corrective Actions

Following a loss of electric power to the ISFSI, plant maintenance personnel will be informed and will isolate the fault and restore service by conventional means. Such an operation is straightforward and routine for the maintenance personnel of an electric utility.

11.1.1.5 Radiological Impact from Off-Normal Operations

No radiological impact from off-normal operations is postulated.

11.1.2 Cask Seal Leakage or Leakage of the Overpressure Monitoring System

The storage casks feature redundant seals in conjunction with an extremely rugged body design. Additional barriers to the release of radioactivity are presented by the sintered fuel pellet matrix and the zircaloy cladding which surrounds the fuel pellets. Furthermore, the interseal gaps are pressurized in excess of the cask cavity. As a result, no credible mechanisms that could result in leakage of radioactive products have been identified. However, to bound the worst off-normal event, leakage of one seal is evaluated.

11.1.2.1 Postulated Cause of the Event

A combined event of failure of one of the seals in addition to a failure of the pressure monitoring system is assessed. This could also be a failure of the pressure boundary of the overpressure system in addition to a failure of the monitoring alarm system.

11.1.2.2 Detection of Event

Detection of a seal leak in addition to the loss of the pressure monitoring system would be by means of periodic testing or maintenance.

11.1.2.3 Analysis of Effects and Consequences

Analysis has been performed in Chapter 3 to show that the lid bolts will prevent leakage of the seals during normal and postulated accident events. A description of the three possible leaks which could occur is presented below:

- In any of the inner confinement seals (lid seal, inner vent seal or inner drain seal).

  The lid and lid penetration cover bolts and seals are designed to prevent leakage during all normal, off-normal and postulated accident events. Therefore this is a very unlikely event.

  In this case the overpressure system, which has a higher pressure than the cask cavity, would leak helium into the cask cavity. Since the pressure is higher in the overpressure tank, it would prevent leakage of radioactive materials out of the cask cavity until the pressure between the overpressure tank and the cask cavity equalized. This would take several years, depending on the size of the leak. At the test leak rate, the overpressure system pressure would always exceed the cask cavity pressure, as shown in Chapter 7. Therefore no leakage of radioactive material can occur, even if the alarm were to fail. Chapter 7 also demonstrates that even if the inner seal has experienced a latent seal failure there is ample time for identifying the leak through routine surveillances.

- In any of the outer seals (lid, overpressure port cover, vent cover or drain cover)

  The lid and lid penetration cover bolts and seals are designed to prevent leakage during all normal, off-normal and postulated accident events. Therefore this is a very unlikely event.

  In this case, leakage out of the interspace to the atmosphere would occur. This would not result in release of radioactive material from the cask cavity since the inner seal is intact. Again, as demonstrated in Chapter 7, a latent seal failure of the outer seals would not result in a release of any radioactive material to the environment. There is also ample time for identifying the leak through routine surveillances.

- A leak in the overpressure system

  This is the most likely cause of a leak, since it is a non safety related component and not designed to withstand accident loadings.

  In this case two scenarios could exist:

  - The overpressure system is not functioning and the inner seal is intact. In this case there is no release of radioactive material to the environment; or

  - The overpressure system is not functioning and the inner seal is leaking at some rate.

In this latter case, leakage out of the interspace to the atmosphere and the cask cavity would occur. This would not result in release of radioactive material from the cask cavity until the pressure fell to the cask cavity pressure. At the test leak rate of 1×10^{-5} ref cm^3/s, this would not occur during the 20 year storage period.

However, a leak of this magnitude in combination with a loss of the over pressure system has been evaluated as both an off-normal and accident condition in Section 7.3.

The results of these calculations assuming off-normal conditions indicate that an individual located at the site boundary (100 m from the cask) for an entire year would receive an effective dose equivalent of 5.39 mrem, a thyroid dose of 1.07 mrem, a bone surface dose of 17.8 mrem, and a lung dose of 22.5 mrem. These doses are below the regulatory limits of 10 CFR 72.104(a) of 2.5×10^{-1} mSv (25 mrem) to the whole body, 7.5×10^{-1} mSv (75 mrem) to the thyroid and 2.5×10^{-1} mSv (25 mrem) to any other critical organ.

The results of these calculations assuming accident conditions indicated that at the site boundary (100m from the cask), for a 30 day release, the total effective dose equivalent is 175 mrem. The total organ dose equivalent to any individual organ (the critical organ in this case is the bone surface) is 927 mrem for a 30 day release. The lens dose equivalent to the lens of the eye is 176 mrem for a 30 day release. These values are well below the limiting off site doses defined in 10 CFR 72.106(b).

Another accident condition under consideration is that the overpressure system is not functioning and the inner seal has experienced a latent seal failure. This analysis is presented in 7.3.3. This accident analysis demonstrates that a latent failure up to 100 times greater that the test value could occur and there is ample time for recovery before the limiting off site doses in 10 CFR 72.106(b) are met. The probability that a gross leak of an inner seal in combination with a gross leak in the outer seal is not considered a credible event.

11.1.2.4 Corrective Action

The overpressure system leak would be repaired at the ISFSI depending on the complexity of the repair, or the cask would be returned to the spent fuel pool for seal replacement. Repairs which could be performed at the ISFSI are tightening of the fittings, replacement of valves or switches, localized weld repairs or replacement of components.

11.1.2.5 Radiological Impact

For the worst case, which includes loss of alarm, complete loss of the pressure differential between the cask and the overpressure system, and complete loss of the overpressure system pressure boundary, the dose rates at the site boundary would increase as stated above, but are below the regulatory limits of 10 CFR 72.104(a).

11.1.3 Overpressure Tank Needs Refilling

The overpressure tank may need to be refilled during the cask storage period to ensure that a positive pressure differential is maintained between the overpressure system and the cask cavity. This maintenance will be performed by plant personnel as scheduled maintenance.

11.1.3.1 Postulated Cause of the Event

Slow leakage of the outer or inner seals, less than the allowable leak rate.

11.1.3.2 Detection of Event

Pressure monitoring system alarm would indicate that the pressure in the overpressure tank had fallen below the set point. The set point is generally set much higher than the maximum cavity pressure so that there is sufficient time to repressurize the tank. Calculations performed in Chapter 7 (See Figure 7.1-1) shows that it would take 11 years to reach the alarm setpoint if the cask were leaking at the seal test leakage acceptance rate of 1×10^{-5} ref cm^3/s. Plant maintenance procedures will be developed to ensure that the tank pressure will be verified or repressurized at least once per ten years.

11.1.3.3 Analysis of Effects and Consequences

The overpressure tank may need to be repressurized during the storage period. This event has no safety or radiological consequences because the set point of the pressure monitoring system is selected so that there is ample time to repressurize prior to any leakage out of the cask cavity.

11.1.3.4 Corrective Actions

After repressurizing, the overpressure tank will be checked to ensure no leakage around the fittings. The pressure transducers/switches will be checked for operability.

11.1.3.5 Radiological Impact

Estimated operational doses due to this action is included in Chapter 10.

11.2 Accidents

Accidents are design events of the third and fourth type (Design Events III and IV) as defined in ANSI/ANS 57.9. Design Event III consists of that set of infrequent events that could reasonably be expected to occur during the lifetime of the ISFSI. Design Event IV consists of the events that are postulated because their consequences may result in the maximum potential impact on the immediate environs. Their consideration establishes a conservative design basis for certain systems with important confinement features.

11.2.1 Earthquake

11.2.1.1 Cause of Accident

The design earthquake (DE) is postulated to occur as a design basis extreme natural phenomenon. The cask is evaluated for a safe shutdown earthquake (SSE) of 0.26g horizontal and 0.17g vertical.

11.2.1.2 Accident Analysis

Cask response to a seismic event is evaluated in Section 2.2.3 and Appendix 3A. Results of these analyses show that the cask does not tip over or slide and that the containment vessel stresses resulting from the seismic loads are below ASME code allowable stresses for accident conditions. The leak-tight integrity of the cask is not compromised. No damage to the cask is postulated. The basket stresses are also low and do not result in deformations that would prevent fuel from being unloaded from the cask.

11.2.1.3 Accident Dose Calculations

The DE does not damage the cask. Hence, no radioactivity is released and there is no associated dose increase due to this event.

11.2.1.4 Corrective Actions

After a seismic event, the cask would be inspected for damage. Any debris would be removed. An evaluation would be performed to determine if the cask were still within the licensed design basis. The pressure monitoring system would be tested, and repaired if necessary. If necessary, the cask would be returned to the spent fuel pool for unloading.

11.2.2 Extreme Wind and Tornado Missiles

11.2.2.1 Cause of Accident

The extreme winds due to passage of the design tornado as defined in Section 2.2.1 are postulated to occur as an extreme natural phenomenon.

11.2.2.2 Accident Analysis

In section 2.2.1, it is shown that extreme winds do not result in a cask tip over or sliding of the cask. The pressure due to high winds on the surface of the cask is bounded by the assumed external pressure of 25 psi. The stresses in the cask resulting from this external pressure are presented in Appendix 3A. High winds have no effect on the leak tight integrity of the cask, and do not result in damage to the cask. High winds do not affect the basket or the ability to retrieve the spent fuel from the cask. The effect of tornado missiles hitting the cask has been evaluated in Section 2.2.1. These analyses show that the stresses in the cask as a result of missile impact are well below the ASME Code allowable stresses for Accident (Level D) conditions. It is also shown in Section 2.2.1 that the tornado missile impact will not result in cask tipover. Local damage to the neutron shield may result from the tornado missile impact. The cask may slide about 7.3 inches due to missile impact below the CG of the cask, however, the space between the two casks is more than 90 inches, therefore, the cask will not impact each other. Table 5.1-2 provides the surface dose rates of the cask assuming that the neutron shield is completely removed. This data can be used by the site to conservatively determine the maximum dose rates at the site boundary due to tornado missile impact.

11.2.2.3 Accident Dose Calculations

Extreme winds are not capable of overturning the casks nor of damaging the cask seals. The overpressure system and the neutron shielding may be damaged. To determine the bounding dose, loss of neutron shielding (Section 11.2.5.3) is combined with the total effective dose equivalent (TEDE) from the loss of one confinement barrier and 100% fuel cladding failure (Section 11.2.9.3). The resulting site boundary accident dose, 888 mrem, is below the 5 rem TEDE limit as specified in 10 CFR 72.106(b).

11.2.2.4 Corrective Actions

After excessive high winds or a tornado, the cask would be inspected for damage. Any debris would be removed. Any damage resulting from impact with a missile would be evaluated to determine if the cask were still within the licensed design basis. The pressure monitoring system would be tested, and repaired if necessary. If necessary, the cask would be returned to the spent fuel pool for unloading.

11.2.3 Flood

11.2.3.1 Cause of Accident

Natural event.

11.2.3.2 Accident Analysis

The postulated floods and high water levels are discussed in Section 2.2.2. The analysis presented shows that the cask will withstand the external pressure due to the flood and the velocity of the flowing water will not result in a cask tip or cause the cask to slide.

11.2.3.3 Accident Dose Calculations

The probable maximum flood is not capable of overturning the casks or of damaging their seals. The overpressure system and the neutron shielding may be damaged. To determine the bounding dose, loss of neutron shielding (Section 11.2.5.3, 713 mrem) is combined with the total effective dose equivalent (TEDE, 175 mrem) from the loss of one confinement barrier and 100% fuel cladding failure (Section 11.2.9.3). The resulting site boundary accident dose, 888 mrem, is below the 5 rem TEDE limit as specified in 10 CFR 72.106(b).

11.2.3.4 Corrective Actions

After a flood of the ISFSI site, the casks would be inspected for damage. The surfaces of the cask would potentially need to be cleaned and repainted in local areas. Any debris would be removed. If there were any damage, an evaluation would be performed to determine if the cask were still within the licensed design basis.

11.2.4 Explosion

11.2.4.1 Cause of Accident

Explosion in the general vicinity.

11.2.4.2 Accident Analysis

Regulatory Guide 1.91provides guidance for Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants. This document states that an acceptable method for demonstrating that a nuclear power plant has the ability to withstand the possible effects of explosions occurring on transportation routes is to demonstrate that the rate of exposure to a peak positive incident overpressure in excess of 1 psi is less than 10^{-6} per year. Further a review of several utility ISFSI Safety Analysis Reports indicate that explosions of a few psi or less are postulated as the worst case explosion.

The TN-68 cask is a robust steel design, and an explosion would not be expected to damage the cask. For conservatism, the cask is evaluated for an external pressure of 25 psi. The cask body is a thick walled construction and is capable of withstanding very high external loads without collapse.

11.2.4.3. Accident Dose Calculations

The cask will not tip as a result of the postulated pressure wave. Accordingly, no cask damage or release of radioactivity is postulated. Since no radioactivity is released, no resultant dose increase is associated with this event.

11.2.4.4 Corrective Actions

After an explosion in the vicinity of the ISFSI site, the casks would be inspected for damage. The surfaces of the cask would potentially need to be cleaned and repainted in local areas. Any debris would be removed. If there were any damage, an evaluation would be performed to determine if the cask were still within the licensed design basis.

11.2.5 Fire

11.2.5.1 Cause of Accident

Combustible materials will not normally be stored at an ISFSI. Therefore, a credible fire would be very small and of short duration such as that due to a fire or explosion from a vehicle or portable crane.

However a hypothetical fire accident is evaluated for the TN-68 cask based on a fuel fire, the source of fuel being that from a ruptured fuel tank of the cask transporter tow vehicle. The bounding capacity of the fuel tank is 200 gallons and the bounding hypothetical fire is an engulfing fire around the cask.

11.2.5.2 Accident Analysis

The evaluation of the hypothetical fire event is presented in Section 4.4 of the SAR. The fire thermal evaluation is performed primarily to demonstrate the containment integrity of the TN-68. This is assured as long as the metallic lid seals remain below 536°F and the cavity pressure is less than 100 psig.

Based on the thermal analyses for the fire accident conditions, the TN-68 packaging can withstand the hypothetical fire accident event without compromising its containment integrity. No melting of the metallic cask components occurs. Peak cask component temperatures are summarized in Table 4.4-1. The maximum seal temperature is calculated to be 485°F which is well below the temperature limit of the metallic seals. The average cavity gas temperature peaks at 572°F and the pressure increases to 71.7 psig (5.88 atm abs). See Section 4.7.5. The pressure inside the cask cavity is well below the design pressure of 100 psig.

The neutron shield will off-gas during the hypothetical accident. A pressure relief valve is provided on the outer shell to prevent the pressurization of the outer shell. Shielding analyses have been performed showing acceptable consequences even if all the resin disappears. (See Chapter 5).

11.2.5.3 Accident Dose Calculations

Local damage to the neutron shielding may result from the fire. This is bounded by removal of all the neutron shielding which is evaluated in Chapter 5. Even with this conservative assumption, the site boundary accident dose rates are below 5 rem to the whole body or any organ as specified in 10CFR72.106(b).

The off-site doses are evaluated for two accident conditions:

1) loss of radial neutron shielding
2) loss of the protective cover and top neutron shield.

For accident conditions, the following assumptions are made:

a) the nearest postulated site boundary is 100 meters distant from the cask
b) the accident involves a single cask
c) the accident duration is 30 days
d) a person remains at the postulated site boundary 24 hours per day for the entire duration

The normal condition direct dose rates at 100 meters are scaled by the ratio of accident to normal surface dose rates as shown in the following table. All units are mrem/hr.

	normal dose rate 100 m Table 5.1-3	accident, surface Table 5.1-2	normal, surface Table 5.1-2	accident, 100 m mrem/hr
gamma	4.0562E-02	774	98.0	3.204E-01
neutron	6.7362E-03	2032	20.6	6.677E-01
			Total:	9.881E-01

The direct dose over 30 days would be 711 mrem. The background from the rest of the ISFSI would be 1/12 of the 25 mrem/year limit (10 CFR 72.104), or 2 mrem. The combined total accident dose would be 713 mrem.

11.2.5.4 Corrective Actions

After a fire, the cask would be inspected for damage. The surfaces of the cask would potentially need to be cleaned and repainted in local areas. The neutron shielding material may have been damaged during the fire. If there is any damage, an evaluation would be performed to determine if the cask were still within the licensed design basis. If the cask is no longer within the design basis, the cask will be returned to the spent fuel pool and unloaded. The neutron shield may need to be replaced prior to putting the cask back into service.

11.2.6 Inadvertent Loading of a Newly Discharged Fuel Assembly

11.2.6.1 Cause of Accident

The possibility of a spent fuel assembly, with a heat generation rate greater than 0.441 kW, being erroneously selected for storage in a cask has been considered. The cause of this accident is postulated to be an error during the loading operations, e.g., wrong assembly picked by the fuel handling crane, or a failure in the administrative controls governing the fuel handling operations.

11.2.6.2 Accident Analysis

The fuel assemblies require several years of storage in the spent fuel pool before the heat generation decays to a rate below 0.441 kW. In addition, the shielding analysis assumes that the fuel has been cooled at least 7 years prior to loading. This accident scenario postulates the inadvertent loading of an assembly not intended for storage in the cask, with a heat generation rate in excess of the design basis specified in Section 2.1.

In order to preclude this accident from going undetected, and to ensure that appropriate corrective actions can take place prior to the sealing of the casks, a final verification of the assemblies loaded into the casks and a comparison with fuel management records is required to assure that the correct assemblies are loaded.

These administrative controls and the records associated with them will be included in the procedures described in Chapter 8.

Appropriate and sufficient actions will be taken to ensure that an erroneously loaded fuel assembly does not remain undetected. In particular, the storage of a fuel assembly with a heat generation in excess of 0.441 kW is not considered credible in view of the multiple administrative controls. Also, surface radiation dose measurements will provide final verification that a newly discharged fuel assembly has not been loaded in the cask.

There is no thermal or shielding analysis impact since the improperly loaded cask will not get out of the water due to independent review. Criticality is not a concern provided that the initial enrichment limit is not exceeded. The loading of a higher enriched fuel assembly is evaluated as a separate accident in section 11.2.7.

11.2.6.3 Accident Dose Calculations

The inadvertent loading of a newly discharged fuel assembly not intended for storage is prevented by administrative control. Therefore, no resultant radiation dose increases would occur.

11.2.6.4 Corrective Actions

If it is determined that a fuel assembly has been loaded which is outside the bounds of the design basis, it shall be removed from the cask.

11.2.7 Inadvertent Loading of a Fuel Assembly with a higher initial enrichment than the Design Basis Fuel

11.2.7.1 Cause of Accident

The possibility of a spent fuel assembly with initial enrichment greater than permitted by the Technical Specifications has been considered. The cause of this accident is postulated to be an error during the loading operations, e.g., wrong assembly picked by the fuel handling crane, or a failure in the administrative controls governing the fuel handling operations.

11.2.7.2 Accident Analysis

This accident is prevented by administrative controls specified in the operations in Chapter 8. Prior to loading of the fuel, the basket type must be verified by the cask serial number, and the pre-selected fuel must be checked to verify that each bundle's enrichment is at or below the limit specified for that basket type.

In order to preclude this accident from going undetected, and to ensure that appropriate corrective actions can take place prior to the sealing of the casks, a final verification of the assemblies loaded into the casks and a comparison with fuel management records is required to assure that the correct assemblies are loaded.

These administrative controls and the records associated with them will be included in the procedures described in Chapter 8.

Appropriate and sufficient actions will be taken to ensure that an erroneously loaded fuel assembly does not remain undetected.

11.2.7.3 Accident Dose Calculations

The inadvertent loading of a fuel assembly with higher initial enrichment than the design basis is prevented by administrative control.

The criticality safety evaluations provide a large margin of conservatism by the assumption that the fuel is unirradiated and contains no burnable poison. The 0.95 limit on k_{eff} for normal, off-normal, and credible accident conditions provides further safety margin. An evaluation of loading a 5% enriched 10x10 fuel assembly at the center of the basket designed for 3.7% enrichment shows that k_{eff} increases from 0.9221 to 0.9260, a change of 0.004, less than 10% of the 0.05 safety margin (Table 6.4-3). Therefore, in the event that a fuel assembly with higher initial enrichment were loaded, the fuel would remain well below critical.

There is no resultant dose rate increase due to this condition.

11.2.7.4 Corrective Actions

If it is determined that a fuel assembly has been loaded which is outside the bounds of the design basis, it shall be removed from the cask.

11.2.8 Hypothetical Cask Drop and Tipping Accidents

11.2.8.1 Cause of Accident

The stability of the TN-68 storage cask in the upright position on the ISFSI concrete storage pad is demonstrated in Section 2.2 of this SAR. The effects of tornado wind and missiles, flood water and earthquakes are described in Sections 2.2.1, 2.2.2 and 2.2.3, respectively. It is shown in those sections that the cask will not tip over under the most severe natural phenomena specified in this Topical Safety Analysis Report.

The cask is designed for single failure proof lifting at the reactor site.

An 18 inch vertical cask drop is postulated to occur during handling while the cask is moved onto or off of a transport vehicle. The trunnions are designed to the requirements of ANSI N14.6[(2)] for lifting devices. The cask will generally be handled by a transport vehicle in a vertical orientation and not lifted higher than 18 in. Other drop events which may be postulated at a specific ISFSI site will be evaluated in accordance with 10 CFR 72.212.

This section of the SAR considers design events of the third and fourth types (includes accidents) as defined in ANSI/ANS 57.9. The third type of events are those that could reasonably be expected to occur over the lifetime of the ISFSI (does not include tipping of the cask). The fourth type of events include severe natural phenomena (described in Section 11.2.1 through 11.2.5) and man induced low probability events postulated because their consequences could result in the maximum potential impact on the immediate environs. Therefore the cask is examined for both dropping and tipping accidents, which are hypothetical impact events that are extremely unlikely to occur.

11.2.8.2 Accident Analyses

The cask is evaluated under bottom end impact on the ISFSI storage pad after a drop from a height of 18 inches in Section 3A.2.3.2. The storage pad is the hardest concrete surface outside of the spent fuel storage building. The cask is generally oriented vertically and not lifted higher than 18 in. once it leaves the containment building. Therefore this case is an upper bound drop event since impact onto a softer surface would result in lower cask deceleration and a lower impact force. The cask is also evaluated under a tipover event on the storage pad even though (as demonstrated in Section 2.2) the cask can not tip over. Appendix 3D determines the maximum g loading which would result for a cask end drop and a tipover. The maximum deceleration due to an 18 inch bottom end drop is 55.5 g's. The maximum deceleration due to a tipover accident is 65 g's.

The cask is analyzed conservatively for an 60 g vertical load simulating the end drop, and a 65g side drop conservatively simulating the tipover. The analyses are presented in Section 3A.2.3.2.

The cask stresses for the cask tip over and drop event are reported in Tables 3A.2.5-15 through -26. All stresses meet the design criteria. An additional analysis of a cask tipping over and impacting on the trunnions is evaluated in 3A.2.4.3. This analysis shows that the local stresses around the trunnion are acceptable, and the g loadings are less severe than the side drop analyzed in 3A.2.3.2.

The stresses in the lid bolts due to the two postulated drop accidents are presented in Section 3A.3.2. This analysis shows that the stresses in the bolts due to the accident loads are well below the allowable limit of $3S_m$ and the bolt yield strength.

The stresses in the basket due to the two postulated drop accidents are presented in Appendix 3B. These analyses show that the basket is structurally satisfactory under the tipover and end drop loads.

Depending on site constraints and requirements, there may be handling conditions different than those analyzed above. For example, there may be a need to lift the cask higher than 18 inches over an impact limiter or a surface which is softer than the ISFSI concrete pad. Prior to using the cask at these sites, 10 CFR 72.212 evaluations shall be performed to ensure that the g loading on the cask is bounded by the g loadings presented above.

11.2.8.3 Accident Dose Calculations

Cask tip will not breach the cask confinement barrier. No radioactivity will be released and no resultant doses will occur.

To determine the bounding dose, the loss of neutron shielding (Section 11.2.5.3) is combined with the total effective dose equivalent (TEDE) from the loss of one confinement barrier and 100% fuel cladding failure (Section 11.2.9.3). The resulting site boundary accident dose, 888 mrem, is below the 5 rem TEDE limit as specified in 10 CFR 72.106(b).

This conservatively bounds any damage to the neutron shield as a result of a cask tip over or drop event.

11.2.8.4 Corrective Actions

After a tipover or cask handling drop, the cask would be inspected for damage. The neutron shielding material may have been damaged due to impact. If there is any damage, an evaluation would be performed to determine if the cask were still within the licensed design basis. If the cask is no longer within the design basis, the cask will be returned to the spent fuel pool and unloaded. The neutron shield may need to be replaced prior to putting the cask back into service.

11.2.9 Loss of Confinement Barrier

11.2.9.1 Cause of Accident

It is assumed that the overpressure system has stopped functioning and fire conditions exist.

11.2.9.2 Accident Analysis

It is assumed that at least one set of seals is still functioning, and that material can be released at the test leak rate of 1×10^{-5} ref cm^3/s. It is also assumed that all of the fuel rods have failed, and the temperature inside the cask is comparable to the fire accident conditions. The cask is assumed to leak at this rate for 30 days.

In this accident, the confinement function of the fuel rod cladding and one set of seals is eliminated. Heat removal and radiation shielding functions operate in the normal passive manner.

This is equivalent to breaking one cask seal barrier, removing the pressure monitoring system, failing all the cladding in all the loaded fuel assemblies (gap activity release), and finally, failing the fuel pellets themselves. The analysis is presented in Section 7.3.2.

11.2.9.3 Accident Dose Calculations

The dose evaluation due to this postulated accident is given in Section 7.3.2.1. The total effective dose equivalent is 175 mrem. The total organ dose equivalent to any individual organ (the critical organ in this case is the bone surface) is 927 mrem for a 30 day release. The lens dose equivalent to the lens of the eye is 176 mrem for a 30 day release. The shallow dose equivalent to the skin is 1.67 mrem/30 days. These values are well below the limiting off site doses defined in 10 CFR 72.106.

11.2.9.4 Corrective Actions

In the event of cask leakage, the cask would be returned to the spent fuel pool and the seals would be replaced. In addition the overpressure system would be checked to determine the cause of failure and corrective measures to prevent future recurrence would be taken. The overpressure system and pressure monitoring equipment would be repaired or replaced as necessary prior to returning the loaded cask to the ISFSI for storage.

11.2.10 Buried Cask

11.2.10.1 Cause of Accident

Earthquake or other natural phenomenon resulting in collapse of building, other structure or other manmade or earthen material onto a cask.

11.2.10.2 Accident Analysis

An evaluation was made to determine the increase in cask temperature with time assuming the cask was completely buried in a medium which will not provide the equivalent cooling of natural

convection and unrestricted radiation to the environment. The details of this analysis are provided in Section 4.4.

The results of this analysis show that if the cask is not uncovered shortly after the accident, the neutron shield temperature will exceed the allowable long term temperature limit of 300°F (149°C). The cavity pressure, including the contribution due to 100 % fuel failure, will exceed 100 psig at approximately 73 hours. The cask seal temperature will not reach its 536°F (280°C) limit at this time. The fuel temperature off-normal limit of 1058°F (570°C) is reached about 87 hours after burial occurs.

11.2.10.3 <u>Accident Dose Calculations</u>

Slow degradation of the neutron shielding would begin to occur shortly after burial resulting in higher surface dose rates. At about 73 hours, the cask internal pressure exceeds the design pressure of 100 psig. The seals will not reach their long term maximum temperature of 536°F (280°C) even 120 hours after burial occurs (see Table 4.4-2). In the event that the cask could not be unburied after 73 hours, release of radioactive gases could occur.

To determine the bounding dose, loss of neutron shielding (Section 11.2.5.3) is combined with the total effective dose equivalent (TEDE) from the loss of one confinement barrier and 100% fuel cladding failure (Section 11.2.9.3). The resulting site boundary accident dose, 888 mrem, is below the 5 rem TEDE limit as specified in 10 CFR 72.106(b).

11.2.10.4 <u>Corrective Actions</u>

The cask should be unburied as soon as possible to prevent release of radioactive material. The cask will be inspected for damage. The neutron shielding material may have been damaged during the burial. If there is any damage, an evaluation would be performed to determine if the cask were still within the licensed design basis. If the cask is no longer within the design basis, the cask will be returned to the spent fuel pool and unloaded. The neutron shield and all seals would need to be replaced prior to putting the cask back into service.

11.3 REFERENCES

1. American Nuclear Society, ANSI/ANS-57.9, Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type), 1992.

2. American National Standards Institute, ANSI N14.6, Special Lifting Devices for Shipping Containers Weighing 10,000 pounds or More, 1986.

CHAPTER 12

OPERATING CONTROLS AND LIMITS

The Technical Specifications for the TN-68 system are included in Appendix A to TN-68 Certificate of Compliance No. 1027, Amendment No. 1.

The Technical Specifications Bases are included herein.

TABLE OF CONTENTS

B 2.0 FUNCTIONAL AND OPERATIONAL LIMITS

B 2.1.1 Fuel to be Stored in the TN-68 Cask

BASES

BACKGROUND	The cask design requires certain limits on spent fuel parameters, including fuel type, assembly weight, initial enrichment, maximum burnup, minimum cooling time prior to storage in the cask, and physical condition of the spent fuel to safely store the spent fuel in the cask. These limitations are included in the thermal, structural, radiological and criticality evaluations performed for the cask.
APPLICABLE SAFETY ANALYSIS	Various analyses have been performed that use these spent fuel parameters as assumptions. These assumptions are included in the thermal, criticality, structural, shielding and confinement analyses. The fuel geometry is determined by the fuel type designation (i.e. GE4, GE5, etc). The maximum uranium content is not generally specified for each fuel type. However, the fuel manufacturer is required to provide the uranium content for each assembly. The shielding analysis is conservatively based on a uranium content either greater than or equal to the TS value for uranium content. The user verification of fuel parameters may be done by administrative review. It is recognized that rod pitch, rod outside diameter and channel thickness values are design nominal values. The limitations for the storage of damaged fuel are based on structural analysis demonstrating that damaged fuel will be retrievable under normal and off-normal conditions. Technical Specification Table 2.1.1-1 provides for minimum cooling times based on a fuel minimum initial enrichment and maximum burnup for 7x7 fuel. To use the table, the minimum enrichments are rounded down and the burnups are rounded up. For example, fuel with a 2.68% enrichment and a burnup of 34.2 GWd/MTU would use the 2.6% enrichment row and the 35 GWd/MTU column. For 8x8, 9x9, and 10x10 fuel, the same function is fulfilled by a fuel qualification flowchart and decay heat formula.
FUNCTIONAL AND OPERATIONAL LIMITS	The Functional and Operational Limits are established to protect the integrity of the fuel clad barrier and the public from radioactive materials in effluents and direct radiation levels associated with cask operation.

(continued)

BASES

FUNCTIONAL AND OPERATING LIMITS VIOLATIONS

2.2.1 If Functional and Operating Limit 2.1 is violated, the limitations on the fuel assemblies in the cask have not been met. Actions must be taken to place the affected fuel assemblies in a safe condition. This safe condition may be established by returning the affected fuel assemblies to the spent fuel pool. However, it is acceptable for the affected fuel assemblies to remain in the cask if that is determined to be a safe condition.

2.2.2 and 2.2.3

Notification of the violation of a Functional and Operating Limit to the NRC is required within 24 hours. Written reporting of the violation must be accomplished within 30 days. This notification and written report are independent of any reports and notification that may be required by 10 CFR 72.75.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1, 3.0.2, 3.0.4, and 3.0.5 establish the general requirements applicable to all Specifications in Sections 3.1 and 3.2 and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the cask is in the specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that: a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified. There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore equipment or variables to within specified limits. Whether stated as a required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS. The second type of Required Action specifies the remedial measures that permit continued operation that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation. Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

(continued)

BASES

LCO 3.0.2 (continued)	The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed. When a change in specified condition is required to comply with Required Actions, the cask may enter a specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable and the ACTIONS Condition(s) are entered.
LCO 3.0.3	This specification is not applicable to a cask. The placeholder is retained for consistency with the power reactor technical specifications.
LCO 3.0.4	LCO 3.0.4 establishes limitations on changes in specified conditions in the Applicability when an LCO is not met. It precludes placing the cask in a specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist: a. Conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the cask being required to exit the Applicability desired to be entered to comply with the Required Actions. Compliance with Required Actions that permit continued operation of the cask for an unlimited period of time in a specified condition provides an acceptable level of safety for continued operation. Therefore, in such cases, entry into a specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring equipment or variables to within specified limits before entering an associated specified condition in the Applicability.

BASES

LCO 3.0.4 (continued)	The provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of a cask. Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification. Surveillances do not have to be performed on the associated equipment out of service (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing specified conditions while in an ACTIONS Condition, either in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or 3.0.4 for those Surveillances that do not have to be performed due to the associated out of service equipment.
LCO 3.0.5	LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate: a. The equipment being returned to service meets the LCO; or b. Other equipment meets the applicable LCOs. The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance.
LCO 3.0.6	This specification is not applicable to a cask. The placeholder is retained for consistency with the power reactor technical specifications.
LCO 3.0.7	This specification is not applicable to a cask. The placeholder is retained for consistency with the power reactor technical specifications.

B 3.0 SURVEILLANCE REQUIREMENTS (SR) APPLICABILITY

BASES

SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications in Sections 3.1 and 3.2 and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirements that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify that equipment and variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to meet the LCO when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components meet the associated LCO when:

a. The systems or components are known to not meet the LCO, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the cask is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on equipment that has been determined to not meet the LCO because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to service.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment within its LCO. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current specified conditions in the Applicability due to the necessary cask parameters not having been established. In these situations, the equipment may be considered to meet the LCO provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a specified condition where other necessary post maintenance tests can be completed.

(continued)

BASES

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. Therefore, when a test interval is specified in the regulations, the test interval cannot be extended by the TS, and the SR includes a Note in the Frequency stating, "SR 3.0.2 is not applicable".

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment as not meeting the LCO or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

(continued)

BASES

SR 3.0.3 (continued)

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a frequency based not on time interval, but upon specified conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of changes in the specified conditions in the Applicability imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment does not meet its LCO Conditions, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

(continued)

BASES

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability. This Specification ensures that equipment requirements and variable limits are met before entry into specified conditions in the Applicability for which this equipment ensures safe operation of the cask.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring equipment to an appropriate status before entering an associated specified condition in the Applicability.

However, in certain circumstances failing to meet an SR will not result in SR 3.0.4 restricting a change in specified condition. When a system, subsystem, division, component, device, or variable is outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that Surveillances do not have to be performed on equipment outside of specified limits. When equipment is outside specified limits, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restriction that may (or may not) apply to specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of a cask.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required(to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

B 3.1 CASK INTEGRITY

B 3.1.1 Cask Cavity Vacuum Drying

BASES

BACKGROUND

A cask is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Functional and Operational Limits. A lid is then placed on the cask. Subsequent operations involve moving the cask to the decontamination area and removing water from the cask fuel cavity. After the cask lid is secured, vacuum drying of the cask cavity is performed and the cavity is backfilled with helium. During normal storage conditions, the cask is backfilled with helium, which is a better conductor than air or vacuum, which results in lower temperatures for stored fuel and the basket.

Cavity vacuum drying is utilized to remove residual moisture from the fuel cavity after the cask has been drained of water. Any water which was not drained from the cask cavity evaporates from fuel or basket surfaces due to the vacuum. This is aided by the temperature increase due to the heat generation of the fuel.

APPLICABLE SAFETY ANALYSIS

The confinement of radioactivity during the storage of spent fuel in a cask is ensured by the use of multiple confinement barriers and systems. The barriers relied upon are uranium dioxide fuel pellet matrix, metallic fuel cladding tubes in which the fuel pellets are contained, and the cask in which the fuel assemblies are stored. Long-term integrity of the fuel cladding depends on storage in an inert atmosphere. This protective environment is accomplished by removing water from the cask cavity and backfilling the cavity with an inert gas. The failure of storage cask confinement capability is considered in the accident analysis (Reference 1).

LCO

A vacuum pressure of less than 4 mbar held for 30 minutes indicates that all liquid water has evaporated and has been removed from the cask cavity. Removing water from the cask cavity helps to ensure the long term minimization of fuel clad corrosion.

APPLICABILITY

Cavity vacuum drying is performed during LOADING OPERATIONS before the cask is transported to the ISFSI storage pad. Therefore, the vacuum requirements do not apply after the cask is backfilled with helium prior to TRANSPORT OPERATIONS and STORAGE OPERATIONS.

(continued)

BASES (continued)

ACTIONS

A.1

The thermal analyses of the cask are performed assuming that helium is in the cask. But during the period from draining of the cask until evacuation of the air and its replacement by helium, heat conduction out of the fuel occurs through air, which has lower conductivity than helium. For the design basis load of 30 kW under air, the maximum fuel cladding temperature will reach the limit of 400°C at 37 hours. If the cavity vacuum drying pressure limit cannot be achieved within 24 hours of completion of cask draining, the cask must be backfilled with helium (a pressure greater than 0.1 atm abs is sufficient to provide required thermal conductivity) within 6 hours. This results in the cask being backfilled with helium within 30 hours of draining the cask. ACTION A.1 requires backfilling with helium to maintain the cask in an analyzed condition, thus allowing additional time to determine the source of the vacuum drying problem.

After the introduction of helium at 30 hours, in the steady state the fuel cladding does not reach the temperature limit and basket thermal expansion is within acceptable limits for a design basis load of 30 kW (Reference 2).

Establishment of even a low pressure helium environment satisfies the helium properties described in design basis thermal analyses because thermal conductivity of gases is not pressure dependent until a high vacuum is attained. Thereby, design basis heat removal requirements will be satisfied over the period of time that it may take to remedy a leaking cask. The near-term effects of not providing a completely dried and pressurized helium atmosphere during this period are negligible. Insignificant corrosion of materials would occur during this period.

Required Action A.1 is modified by a note which allows exiting the LCO in the event that the nominal helium cask environment must be vented during subsequent actions that may be necessary to remedy the condition. For example, the helium may be vented and the LCO exited if it is discovered that residual water must be drained from the cask prior to re-commencing the vacuum drying process.

ACTIONS

A.2

If the cask cavity vacuum drying pressure limit cannot be achieved, actions must be taken to meet the LCO. Failure to successfully complete cavity vacuum drying could have many causes, such as failure of the vacuum drying system, inadequate draining, ice clogging of the drain lines, or leaking of the cask seals. Once the helium atmosphere is established by Action A.1, there is enough conduction to maintain the loaded fuel and the basket within their temperature limits. Therefore, no time limit is required for this action, other than completion prior to final helium backfill.

(continued)

BASES

ACTIONS (continued)

B.1

If a nominal helium environment cannot be achieved or maintained in the cask within 37 hours of draining the cask, the fuel cladding could exceed its temperature limit. Therefore the cask will be placed back into the spent fuel pool within 7 days. Seven days is sufficient time to reflood the cask. Once placed in the spent fuel pool with the lid removed, the fuel is provided with adequate decay heat removal facilities to maintain the loaded fuel within limits.

C.1

If the cask cavity drying pressure limits cannot be achieved within the Completion Time of 7 days, actions must be taken to return the cask to the spent fuel pool within the ensuing 30 days. Evaluation and repair to cask drying equipment may continue. Once placed in the spent fuel pool with the lid removed, the fuel is provided with adequate decay heat removal facilities to maintain the loaded fuel within limits.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

Cavity dryness is demonstrated by evacuating the cavity to a very low pressure and verifying that the pressure is held over a specified period of time. A high vacuum is an indication that the cavity is dry.

This dryness test must be performed successfully on each cask before placing in storage. The test must be performed within 24 hours of draining the cask and removing it from the spent fuel pool. This period allows sufficient time to prepare the cask and perform the test while minimizing the time the fuel is in the cask without a helium atmosphere.

At steady state for a thermal load below 22 kW, the fuel cladding and basket do not reach their temperature limits, so there is no time limit for completion of vacuum drying (Reference 2). At or above 22 kW, the same applies once the air is evacuated and replaced by helium at any time during vacuum drying.

REFERENCES

1. SAR Section 11.1.2, Cask Seal Leakage

2. SAR Section 4.5.1, Vacuum Drying

B 3.1 CASK INTEGRITY

B 3.1.2 Cask Helium Backfill Pressure

BASES

BACKGROUND	A cask is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Functional and Operational Limits. A lid is then placed on the cask. Subsequent operations involve moving the cask to the decontamination area and removing water from the cask fuel cavity. After the cask lid is secured, vacuum drying of the cask cavity is performed, and the cavity is backfilled with helium. During normal storage conditions, the cask is backfilled with helium, which is a better conductor than air or vacuum, which results in lower temperatures for stored fuel and the basket. Backfilling the cask cavity with helium promotes heat transfer from the fuel and the inert atmosphere protects the fuel cladding. Providing a helium pressure greater than atmospheric pressure ensures that there will be no in-leakage of air over the life of the cask.
APPLICABLE SAFETY ANALYSIS	The confinement of radioactivity during the storage of spent fuel in a cask is ensured by the use of multiple confinement barriers and systems. The barriers relied upon are uranium dioxide fuel pellet matrix, metallic fuel cladding tubes in which the fuel pellets are contained, and the cask in which the fuel assemblies are stored. Long-term integrity of the fuel cladding depends on storage in an inert atmosphere. This is accomplished by removing water from the cask cavity and backfilling the cavity with an inert gas. The failure of storage cask confinement capability is considered in the accident analysis (Reference 1). In addition, the thermal analyses of the cask STORAGE OPERATIONS assume that the cask cavity is filled with helium.
LCO	Backfilling the cask cavity with helium at a pressure exceeding atmospheric pressure will ensure that there will be no air in-leakage into the cavity which could damage the fuel cladding over the licensed storage period. The helium pressure of 2.0 atm abs (+0/-10%) was selected to ensure that the pressure within the cask remains within the design pressure limits over the life of the cask. The helium pressure is the as left value immediately after helium fill is completed in preparation for long term storage.
APPLICABILITY	Helium backfill is performed during LOADING OPERATIONS prior to transporting the cask to the ISFSI storage pad. The helium leak rate is then measured prior to TRANSPORT OPERATIONS and STORAGE OPERATIONS.

(continued)

BASES (continued)

ACTIONS

A.1

The thermal analyses of the cask are performed assuming that helium is in the cask. If the cask cavity helium pressure limit cannot be achieved within 30 hours of completion of cask draining, the cask must be backfilled with helium (a pressure greater than 0.1 atm abs is sufficient to provide required thermal conductivity) immediately, i.e., within 30 hours of draining the cask. This time limit is achievable because a helium environment is generally maintained in the cask after vacuum drying. For the design basis load of 30 kW under air, the maximum fuel cladding temperature will reach the limit of 400°C at 37 hours. ACTION A.1 requires backfilling the helium with helium to maintain the cask in an analyzed condition, thus allowing additional time to determine the source of the helium backfill problem.

Establishment of even a low pressure helium environment satisfies the helium properties described in design basis thermal analyses because thermal conductivity of gases is not pressure dependent until a high vacuum is attained. Thereby, design basis heat removal requirements will be satisfied over the period of time that it may take to remedy a leaking cask. The near-term effects of not providing a completely dried and pressurized helium atmosphere during this period are negligible. Insignificant corrosion of materials would occur during this period.

Required Action A.1 is modified by a note which allows exiting the LCO in the event that the nominal helium cask environment must be vented during subsequent actions that may be necessary to remedy the condition.

A.2

If the initial helium backfill pressure cannot be obtained, actions must be taken to meet the LCO. Once the helium atmosphere is established by Action A.1, there is enough conduction to maintain the loaded fuel within its temperature limits. Therefore, no time limit is required for this action, other than completion prior to helium leak testing.

(continued)

BASES

ACTIONS (continued)

B.1

If a nominal helium environment cannot be achieved or maintained in the cask within 37 hours of draining the cask, the fuel cladding could exceed its temperature limit. Therefore the cask will be placed back into the spent fuel pool within 7 days. Seven days is sufficient time to reflood the cask. Once placed in the spent fuel pool with the lid removed, the fuel is provided with adequate decay heat removal facilities to maintain the loaded fuel within limits.

C.1

If the helium backfill limits cannot be achieved, actions must be taken to return the cask to the spent fuel pool within the ensuing 30 days. Evaluation and repair to helium backfill equipment may continue. Once placed in the spent fuel pool with the lid removed, the fuel is provided with adequate decay heat removal facilities to maintain the loaded fuel within limits.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

The long-term integrity of the stored fuel is dependent on storage in a dry, inert environment and maintenance of adequate heat transfer mechanisms. Filling the cask cavity with helium at the initial pressure specified will ensure that there will be no air in-leakage, which could potentially damage the fuel and that the cask cavity internal pressure will remain within limits for the life of the cask.

Backfilling with helium must be performed successfully on each cask before placing in storage. The SR must be performed within 30 hours after draining the cask. This time is limited to ensure that the fuel cladding and basket do not exceed their temperature limits. This 30 hour period is sufficient time to backfill the cask cavity with helium while minimizing the time the fuel is in the cask without the assumed thermally-conductive atmosphere.

Below 22 kW, the fuel cladding and basket will not reach their temperature limits, so there is no time limit for completion of helium backfill (Reference 2). At or above 22kW, the same applies once the air is evacuated and replaced by helium at any time prior to 30 hours from draining.

REFERENCES

1. SAR Section 11.1.2, Cask Seal Leakage
2. SAR Section 4.5.1, Vacuum Drying

B 3.1 CASK INTEGRITY

B 3.1.3 Cask Helium Leak Rate

BASES

BACKGROUND	A cask is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Functional and Operational Limits. A lid is then placed on the cask. Subsequent operations involve removing water from the cask fuel cavity and moving the cask to the decontamination area. After the cask lid is secured, vacuum drying of the cask cavity is performed, and the cavity is backfilled with helium. During normal storage conditions, the cask is backfilled with helium, which is a better conductor than air or vacuum, which results in lower temperatures for stored fuel and the basket. Backfilling the cask cavity with helium promotes heat transfer from the fuel and the inert atmosphere protects the fuel cladding. Prior to moving the cask to the storage pad, the helium leak rate is determined to ensure that the fuel is confined.
APPLICABLE SAFETY ANALYSIS	The confinement of radioactivity during the storage of spent fuel in a cask is ensured by the use of multiple confinement barriers and systems. The barriers relied upon are uranium dioxide fuel pellet matrix, the metallic fuel cladding tubes in which the fuel pellets are contained, and the cask in which the fuel assemblies are stored. Long-term integrity of the fuel cladding depends on storage in an inert atmosphere. This is accomplished by removing water from the cask cavity and backfilling the cavity with an inert gas. The failure of one of the confinement barriers is considered as an off-normal condition (Reference 1). In addition, the thermal analyses of the cask STORAGE OPERATIONS assume that the cask cavity is filled with helium.
LCO	Verifying that the cask cavity is sealed by measuring the helium leak rate will ensure that the assumptions in the normal, off-normal, and accident analyses and radiological evaluations are maintained. The helium leak rate value not to exceed 1 x 10-5 ref cc/sec is used in the confinement analyses (Reference 3). This limit is based on air leakage (ref-cc/sec) which requires conversion from helium leakage as appropriate.
APPLICABILITY	During LOADING OPERATIONS, the helium leak rate is required to be met when all lid bolts have had their final tensioning (torqued). Cask seal integrity is monitored during STORAGE OPERATIONS by LCO 3.1.5, Cask Interseal Pressure.

(continued)

BASES

ACTIONS

A.1

If the helium leak rate limit is not met or is unknown due to unsatisfactory results from SR 3.1.3.1, actions must be taken to meet the LCO. The 7 day Completion Time provides ample time to investigate the source of the leak and reestablish the cask helium leak rate within limit.

B.1

The 30-day Completion Time is based on engineering judgment and operating experience that any credible seal leak within the total 37 day period would not result in significant loss of helium inventory that would affect the heat removal capability of the cask. Even in the event of a significant leak, the cask environment would not be reduced to less than one atmosphere of helium because there is no mechanism to exchange the helium in the cask with external air. Based on operational experience with transport casks, this 30 day Completion Time is sufficient to disconnect the test equipment, vent the cask, return the cask to the spent fuel pool so that examination, repairs, seal exchange, or cask unloading can be performed as appropriate.

Once placed in the spent fuel pool with the lid removed, the fuel is provided with adequate decay heat removal facilities to maintain the loaded fuel within limits.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

A primary design consideration of the cask is that it adequately can contain radioactive material. Measuring the helium leak rate with an appropriate detector demonstrates that the confinement barrier is established and within design assumptions (Reference 2).

Measuring the helium leak rate must be performed successfully on each cask prior to placing it in storage. Once the helium atmosphere is established by SR 3.1.2, there is enough conduction to maintain the loaded fuel within its temperature limits, and to prevent thermal expansion from damaging the basket. Therefore, no time limit is required for this surveillance, other than completion prior to Transport Operations.

REFERENCES

1. SAR Section 11.1.2, Cask Seal Leakage
2. SAR Section 9.1.3, Leak Tests
3. SAR Section 7.3, Confinement Requirements for Hypothetical Accident Conditions

B 3.1 CASK INTEGRITY

B 3.1.4 Combined Helium Leak Rate

BASES

BACKGROUND	A cask is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Functional and Operational Limits. A lid is then placed on the cask. Subsequent operations involve removing water from the cask fuel cavity and moving the cask to the decontamination area. After the cask lid is secured, vacuum drying of the cask cavity is performed, and the cavity is backfilled with helium. During normal storage conditions, the cask is backfilled with helium, which is a better conductor than air or vacuum, which results in lower temperatures for stored fuel and the basket. Backfilling the cask cavity with helium promotes heat transfer from the fuel and the inert atmosphere protects the fuel cladding. Prior to moving the cask to the storage pad, the helium leak rate is determined to ensure that the fuel is confined. The overpressure system provides redundant sealing and a means of monitoring the cask confinement system. The overpressure system may be leak tested prior to moving the cask to the storage pad or within 48 hours of moving the cask to the storage pad.
APPLICABLE SAFETY ANALYSIS	The confinement of radioactivity during the storage of spent fuel in a cask is ensured by the use of multiple confinement barriers and systems. The barriers relied upon are uranium dioxide fuel pellet matrix, the metallic fuel cladding tubes in which the fuel pellets are contained, and the cask in which the fuel assemblies are stored. Long-term integrity of the fuel cladding depends on storage in an inert atmosphere. This is accomplished by removing water from the cask cavity and backfilling the cavity with an inert gas. The failure of one of the cask seals is considered as an off-normal condition (Reference 1). In addition, the thermal analyses of the cask STORAGE OPERATIONS assume that the cask cavity is filled with helium.
LCO	Verifying that the overpressure system is sealed by measuring the helium leak rate will ensure that the assumptions in the normal, off-normal, and accident analyses and radiological evaluations are maintained. The helium leak rate value not to exceed 1 x 10-5 ref cc/sec is used in the confinement analyses (Reference 3). This limit is based on air leakage (ref-cc/sec) which requires conversion from helium leakage as appropriate.
APPLICABILITY	During STORAGE OPERATIONS, the helium leak rate is required to be met within 48 hours of moving the cask to the storage pad. Cask seal integrity is monitored during STORAGE OPERATIONS by LCO 3.1.5, Cask Interseal Pressure.

(continued)

BASES (continued)

ACTIONS

The ACTIONS Table is modified by a note indicating that a separate Condition entry is allowed for each cask. This note is acceptable because the internal environment of one cask is independent of the internal environment of subsequent casks or adjacent casks. The Required Actions for each Condition provide appropriate compensatory actions for each cask not meeting the LCO. Subsequent casks that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the helium leak rate limit is not met or is unknown due to unsatisfactory results from SR 3.1.4.1, actions must be taken to meet the LCO. The 48 hour Completion Time of Required Action A.1 provides ample time to investigate the source of the leak and reestablish the cask helium leak rate within limit.

B.1

The 30-day Completion Time is based on engineering judgment and operating experience that any credible seal leak within the total 32-day period would not result in significant loss of helium inventory that would affect the heat removal capability of the cask. Even in the event of a significant leak, the cask environment would not be reduced to less than one atmosphere of helium because there is no mechanism to exchange the helium in the cask with external air. Because the cask has previously passed SR 3.1.3, failure of SR 3.1.4 would be due to an overpressure system leak rather than a confinement boundary leak. Based on operational experience with transport casks, this 30 day Completion Time is sufficient to disconnect the test equipment and return the cask to the a fuel unloading facility where repairs can be made, the cask can be flooded, and the fuel can be removed as appropriate.

Because the cask will retain its helium, the fuel is provided with adequate decay heat removal facilities to maintain the loaded fuel within limits.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

A primary design consideration of the cask is that it adequately can contain radioactive material. Measuring the helium leak rate with an appropriate detector demonstrates that the confinement barrier is established and within design assumptions (Reference 2).

Measuring the helium leak rate must be performed successfully on each cask prior to placing it in storage. The surveillance must be performed within 48 hours of moving the cask to the storage pad. This 48 hour period allows sufficient time to perform the SR while minimizing the time the fuel is in the cask without verifying that the cask is sealed

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	A note has been added to the surveillance to state that SR 3.1.4.1 may be combined with SR 3.1.3.1. This surveillance allows the leak testing of the overpressure system while on the storage pad. However, the surveillance may be performed with the leak testing of the cask seals while in the spent fuel building.
REFERENCES	1. SAR Section 11.1.2, Cask Seal Leakage 2. SAR Section 9.1.3, Leak Tests 3. SAR Section 7.3, Confinement Requirements for Hypothetical Accident Conditions

B 3.1 CASK INTEGRITY

B 3.1.5 Cask Interseal Pressure

BASES

BACKGROUND	A cask is loaded, dried, and sealed prior to being transported to the ISFSI and placed on a storage pad. The cask is designed with redundant seals to contain the radioactive material. In addition, 10CFR72.122(h)(4) and 10CFR72.128(a)(1) state that the casks must have the capability to be continuously monitored such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. The monitoring systems provide the following features: a. The capability to monitor interseal pressure that will indicate if cask seal integrity is compromised; and b. Local alarms to notify the licensee that potential seal degradation has occurred. It is necessary to verify cask seal integrity at a regular interval. Backfilling the cask cavity with helium promotes heat transfer from the fuel and the inert atmosphere protects the fuel cladding.
APPLICABLE SAFETY ANALYSIS	The confinement of radioactivity during the storage of spent fuel in the cask is ensured by the use of multiple confinement barriers and systems. The barriers relied upon are uranium dioxide fuel pellet matrix, the metallic fuel cladding tubes in which the fuel pellets are contained, and the cask in which the fuel assemblies are stored. Long-term integrity of the fuel cladding depends on storage in an inert atmosphere. This is accomplished by removing water from the cask cavity and backfilling the cavity with an inert gas. The failure of storage cask confinement capability is considered in the accident analysis and the off-normal analysis (References 1, 2, and 3). In addition, the thermal analyses of the cask STORAGE OPERATIONS assume that the cask cavity is filled with helium.
LCO	Verifying cask interseal pressure ensures that the assumptions relating to radioactive releases in the accident analyses and radiological evaluations are maintained. Seal integrity is verified by monitoring interseal pressure indication and alarms.

(continued)

BASES (continued)

APPLICABILITY

Cask interseal pressure verification is performed regularly during STORAGE OPERATIONS to confirm that the cask confinement barriers have not been compromised. During LOADING OPERATIONS the seal integrity is verified prior to moving the cask to the ISFSI storage pads. Verification during TRANSPORT OPERATIONS is not possible as the cask is being moved. However, TRANSPORT OPERATIONS are brief and follow the verification performed during LOADING OPERATIONS and, therefore, does not represent a significant lapse in seal integrity monitoring.

ACTIONS

The ACTIONS Table is modified by a note indicating that a separate Condition entry is allowed for each cask. This note is acceptable because the internal environment of one cask is independent of the internal environment of subsequent casks or adjacent casks. The Required Actions for each Condition provide appropriate compensatory actions for each cask not meeting the LCO. Subsequent casks that do not meet the LCO are governed by subsequent Condition entry and application of Required Actions.

<u>A.1</u>

If a condition is entered due to failure to meet SR 3.1.5.1, an appropriate evaluation shall be performed to investigate the cause of the low pressure condition. The 7 day period is sufficient time to perform an assessment of the condition and make repairs to the overpressure system, change out the pressure switch, if necessary, and reestablish a pressure above 3.0 atmospheres. Reestablishing the pressure above 3.0 atmospheres prevents leakage of radioactive material from the cask cavity. However, if the source of the low pressure is due to a leak greater than analyzed in any cask seal or the overpressure system, the leak should be repaired.

<u>B.1</u>

If it is determined that there is a leakage path in the cask or overpressure system, the repair should be performed in a timely manner. If the interseal pressure has been reestablished to 3.0 atmospheres or above, no leakage of radioactive material from the cask cavity can occur.

The 30 day COMPLETION TIME of REQUIRED ACTION B.1 provides ample time to return the cask to the a fuel unloading facility where repairs can be made, the cask can be flooded, and the fuel can be removed as appropriate.

Even if there is a leak to an inner seal, the cask retains at least one atmosphere of helium, so the fuel is provided with adequate decay heat removal to maintain the loaded fuel within temperature limits.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

After the initial leak testing is successfully performed, the cask overpressure tank pressure is routinely monitored every 7 days. This ensures that no leaks have occurred after initial testing is done. Verification of the pressure exceeding 3.0 atmospheres may be performed using alarms, pressure transducers, or other similar verification methods. Seven days is appropriate, based on the low probability of developing a leak during TRANSFER and STORAGE OPERATIONS.

Cask seal integrity must be verified in accordance with 10CFR72.112(h)(4) and 10CFR72.128(a)(1). The method for verifying seal integrity is to monitor the interseal pressure. Normally, the cask seal integrity is verified using installed instrumentation that alarms or indicates. If this system is not operating on one or more casks, monitoring of seal integrity at each affected cask may be performed by alternative means.

SR 3.1.5.2

Cask seal integrity must be verified in accordance with 10CFR72.122(h)(4) and 10CFR72.128(a)(1). To ensure operability of the interseal pressure monitoring system as a remote indicator during STORAGE OPERATIONS, SR 3.1.5.2 verifies the proper functioning and setpoint of the pressure switch and transducer within 7 days of commencing STORAGE OPERATIONS. This verification is a CHANNEL OPERATIONAL TEST (COT) which exercises the pressure switch by reducing the sensed pressure below the setpoint, and which verifies the accuracy of the trip setpoint within the required range. Full channel calibration over the range of the instrument is not required because the instrument provides no analog indication. Subsequent operability in-service is verified by a COT every 36 months, a reasonable period which addresses the expected drift of the instrument and the reliability of the pressure switch testing.

REFERENCES

1. SAR Section 7.3, Confinement Requirements for Hypothetical Accident Conditions
2. SAR Section 11.1.2, Cask Seal Leakage
3. SAR Section 11.2.9, Loss of Confinement Barrier

B 3.1 CASK INTEGRITY

B 3.1.6 Cask Minimum Lifting Temperature

BASES

BACKGROUND	Minimum temperature limits for cask lifting/movement operations must be observed to avoid the potential for brittle fracture of the cask.
APPLICABLE SAFETY ANALYSIS	The containment vessel and the gamma shielding are fabricated from materials selected for their fracture toughness properties at low temperatures. The fracture toughness evaluation is based on a minimum temperature of -20°F. The cask will generally be at a temperature higher than the ambient temperature due to the heat load of the fuel. However, for conservatism and simplicity, it is recommended that the ambient be selected as the minimum cask movement temperature. It is highly unlikely that any cask movement activity would take place at temperatures below -20°F. Nevertheless, if movement at a temperature below that specified is necessary, calculations (similar to those presented in Chapter 4 of the SAR) or direct measurement may be used to estimate the minimum cask surface temperature for any particular ambient condition.
LCO	The LCO requires that the cask not be lifted or moved if the cask outer surface temperature is below that specified.
APPLICABILITY	This technical specification is applicable during TRANSPORT OPERATIONS.
ACTIONS	<u>A.1</u> If the Surveillance Requirements are satisfied prior to the operation, the ambient temperature limits will be met. If, however, the temperature should decrease below the limit during the operation, the cask must be placed on a stable, qualified surface and the lifting/movement operation must be suspended. If the temperature limit is violated during TRANSPORT OPERATIONS, the cask must first be returned to a safe and secure location. Based on the significant margin provided in the calculation of fracture toughness, it is safe to continue TRANSPORTATION OPERATIONS for a short period if the ambient temperatures decrease a few degrees below the limit. For radiological and security reasons, it would be safer to transport the cask to a safe and secure area, as opposed to immediately suspending the operation and establishing temporary security and radiological controls at some temporary location until the time that ambient temperature increased above the specified value. (continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Prior to TRANSPORT OPERATIONS, the cask outer surface temperature should be verified. This temperature requirement can be met by measuring the ambient temperature prior to transport. Weather forecasts should be considered for the planned period of the TRANSPORT OPERATION.

B 3.2 CASK RADIATION PROTECTION

B 3.2.1 Cask Surface Contamination

BASES

BACKGROUND	A cask is immersed in the spent fuel pool in order to load the spent fuel assemblies. As a result, the surface of the cask may become contaminated with the radioactive material in the spent fuel pool water. This contamination is removed prior to moving the cask to the ISFSI in order to minimize radioactive contamination to personnel or the environment. This allows the ISFSI to be entered without additional radiological controls to prevent the spread of contamination and reduce personnel dose due to the spread of loose contamination or airborne contamination. This is consistent with ALARA practices (Reference 1).
APPLICABLE SAFETY ANALYSIS	The radiation protection measures implemented at the ISFSI are based on the assumption that the exterior surfaces of the cask have been decontaminated. Failure to decontaminate the surfaces of the casks could lead to higher than projected occupational doses.
LCO	Removable surface contamination on the cask exterior surfaces is limited to 1000 dpm/100 cm2 from beta and gamma sources and 20 dpm/100 cm2 from alpha sources. These limits are taken from Ref. 1 and are based on the minimum level of activity that can be routinely detected under a surface contamination control program using direct survey methods. Experience has shown that these limits are low enough to prevent the spread of contamination to clean areas and are significantly less than the levels which would cause significant personnel skin dose.
APPLICABILITY	Verification that the cask surface contamination is less than the LCO limit is performed during LOADING OPERATIONS. This occurs prior to TRANSPORT OPERATIONS and STORAGE OPERATIONS, and CONDITION A is not applicable until the SURVEILLANCE REQUIREMENT (SR3.2.1.1) has been performed. Measurement of the cask surface contamination is unnecessary during TRANSPORT OPERATIONS in preparation for UNLOADING OPERATIONS as surface contamination would have been measured prior to moving the cask to the ISFSI.
ACTIONS	<u>A.1</u> If the removable surface contamination of a cask that has been loaded with spent fuel is not within the LCO limits, action must be initiated to decontaminate the cask and bring the removable surface contamination within limits. The Completion Time requires that the decontamination be completed prior to TRANSPORT OPERATIONS, which will prevent the release of contamination to the environment and the ISFSI.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.2.1.1 This SR verifies that the removable surface contamination on the cask is less than the limits in the LCO. The Surveillance is performed using smear surveys to detect removable surface contamination. The Frequency requires performing the verification once; following cask loading and prior to initiating TRANSPORT OPERATIONS. This Frequency is adequate to confirm that the cask can be moved to the ISFSI without spreading loose contamination, and assumes that the cask will not develop surface contamination during TRANSPORT or STORAGE OPERATIONS. Storage of the fuel in the dry, redundantly-sealed cask eliminates the possibility for leakage of contaminated liquids.
REFERENCES	1. USNRC IE Circular 81-07 dated May 14, 1981, "Control of Radioactively Contaminated Materials."

CHAPTER 13

QUALITY ASSURANCE

This chapter establishes the Quality Assurance (QA) program applied to the design, analysis, fabrication, assembly, and testing of TN-68 Storage System components that are Important to Safety as defined in Section 2.3.1 of this SAR.

All quality-related activities will be controlled under an NRC-approved quality assurance program, meeting the requirements of 10CFR72,[1] Subpart G. The licensee's QA program will be used to control activities performed by the licensee.

TN is responsible for the TN-68 Storage System as discussed in Section 1.3 of this SAR. TN implements its Quality Assurance Program for nuclear quality-related activities. The TN Quality Assurance Program is being invoked by TN to provide uniformity in the 10CFR72, Subpart G, quality program. The TN Quality Assurance Procedures are used to implement the provisions of the TN Quality Assurance Program for the nuclear quality-related activities associated with the TN-68 Storage System.

The TN Quality Assurance Program will be applied to the Important to Safety (10CFR72) components of the TN-68 Storage System and to the associated nuclear quality-related activities. In addition to compliance with 10CFR72, Subpart G, guidance for the TN Quality Assurance Procedures have been taken from Regulatory Guide 7.10[2] and from NUREG/CR-6407[3]. These quality procedures are used to establish the quality category of components, subassemblies, and piece parts according to each item's importance to nuclear safety.

The matrix in Table 13-1 shows the 10CFR72, Subpart G, criteria and the respective sections of the TN Quality Program that address the criteria.

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[1] Title 10, U.S. Code of Federal Regulations, Part 72, (10CFR72), Licensing Requirements for the independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, 1995.

[2] Regulatory Guide 7.10, Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material, U.S. Nuclear Regulatory Commission, June 1974.

[3] NUREG/CR-6407, Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety, U.S. Nuclear Regulatory Commission, February 1996.

Table 13-1 – Quality Assurance Criteria Matrix

10CFR72, Subpart G		TN Quality Assurance Manual
Section	**Criteria**	**Section**
72.142	Organization	1
72.144	Quality Assurance Program	2
72.146	Design Control	3
72.148	Procurement Document Control	4
72.150	Instructions, Procedures, and Drawings	5
72.152	Document Control	6
72.154	Control of Purchased Material, Equipment, and Services	7
72.156	Identification and Control of Material, Parts, and Components	8
72.158	Control of Special Processes	9
72.160	Licensee Inspection	10
72.162	Test Control	11
72.164	Control of Measuring and Test Equipment	12
72.166	Handling, Storage, and Shipping Control	13
72.168	Inspection, Test, and Operating Status	14
72.170	Nonconforming Materials, Parts, or Components	15
72.172	Corrective Action	16
72.174	Quality Assurance Records	17
72.176	Audits	18

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CHAPTER 14

DECOMMISSIONING

14.1 Decommissioning Considerations

The TN-68 cask design features inherent ease and simplicity of decommissioning. At the end of its service life cask, decommissioning could be performed by one of the options listed below:

- Option 1, the TN-68, including spent fuel in storage, could be shipped to either a monitored retrievable storage system (MRS) or a geological repository for final disposal, or

- Option 2, the spent fuel could be removed from the TN-68 cask (either at the utility or at another off site location) and shipped in a DOE approved cask.

The first option does not require any decommissioning of the TN-68 cask. No residual contamination is expected to be left behind on the concrete base pad. The base pad, fence, and periphery utility structures will require no decontamination or special handling after the last cask is removed. The ISFSI pad could be demolished with normal construction techniques.

The second option would require decontamination of the TN-68 cask. The sources of contamination in the interior of the cask would primarily be crud left from the spent fuel pool water or crud from the spent fuel pins. These are expected to be low levels of contamination which could simply be removed with high pressure water spray. After decontamination, the TN-68 cask could either be cut up for scrap or partially scrapped. Any metal activation would be shipped as low level radioactive waste to a near surface disposal facility. For surface decontamination of the TN-68, electropolishing or chemical etching can be used to remove the contaminated surface of the cask.

Cask activation analyses have been performed to quantify the specific activities of the cask materials after years of storage. The following assumptions were made:

- the cask contains 68 7x7 BWR fuel assemblies at 40 GWd/MTU and 10 years cooling, and
- the neutron flux is assumed constant for 40 years.

The activation calculation is performed with the 7x7 fuel source identified in Chapter 5 using the computer code ORIGEN2. The total neutron fluxes are taken from a radial SAS1 (one dimensional) shielding calculation performed with the XSDRN-PM code using source term and radial shielding thicknesses similar to those used for 7x7 fuel in Chapter 5. The SAS1 input file is provided in Section 14.2. The fluxes at the cask centerline, the cavity wall, the neutron shield, and the outer shell are used to irradiate the basket, the body, the lid, the neutron shield, and outer shell and protective cover. The fluxes, material compositions and masses of irradiated material are listed in Table 14.1-1. The ORIGEN2 cross section library for BWR's at a burnup of 27,500 MWD/MTU is used.

The results listed in Table 14.1-2 indicate that after 40 years irradiation and 30 days decay (to eliminate very short lived radionuclides), the total activity is less than 0.071 Ci.

To evaluate the TN-68 cask and basket for disposal, the specific activity of the isotopes listed in Tables 1 and 2 of 10 CFR 61.55 is determined and compared with the limits for Class A waste in those tables.

It is expected that after the application of a surface decontamination method, the radiation levels will be below the acceptable limits of Regulatory Guide 1.86.[1] The results of the calculation, shown in Table 14.1-3, show that activation of TN-68 will be far below the specific activity limits for both long and short lived nuclides for Class A waste. A detailed evaluation will be performed at the time of decommissioning to determine the appropriate mode of disposal.

The procedure for decommissioning the TN-68 is summarized below:

- Remove bolts, weather shield, overpressure monitoring system, top neutron shield (polyethylene disc), port covers, quick disconnect fittings, and seals. Evaluate surface contamination and determine if these items should be disposed of as non-radioactive waste or as low-level radioactive waste.

- Wash down the TN-68 basket inside the cask. Pump out and filter contaminated water and cleaning agent.

- Remove basket and rails and wash down again. Cut and crush basket for disposal as low level radioactive waste.

- Decontaminate the lid and basket rails until able to dispose of as scrap metal. If unable to achieve these levels, cut and dispose of as low level radioactive waste.

- Decontaminate the cask body. Cut the outer neutron shield shell and remove the neutron shield boxes. These are not expected to be contaminated; verify and dispose of as non-radioactive waste.

- Verify status of the cask body. It is expected that surface decontamination will be adequate, if so then dispose of the cask body as scrap metal. If unable to decontaminate to these levels, the cask body can be cut and disposed of as low level radioactive waste.

As stated earlier under option 1, due to the leak tight design of the storage casks, no residual contamination is expected to be left behind on the concrete base pad. No special techniques are necessary to remove the concrete pad.

The volume of waste material produced incidental to ISFSI decommissioning is expected to be limited to that necessary to accomplish surface decontamination of the casks if the spent fuel elements must be removed. Furthermore, it is estimated that the cask materials will be slightly activated as a result of their long term exposure to the relatively small neutron flux emanating from the spent fuel, and that the resultant activation level will be well below the allowable limits

for general release of the casks as noncontrolled material. Therefore, it is anticipated that the casks, may be decommissioned from nuclear service by surface decontamination alone, which could be performed at the utility.

A detailed decommissioning plan will be submitted prior to the commencement of decommissioning activities. The costs of decommissioning the ISFSI are expected to represent a small and negligible fraction of the cost of decommissioning a nuclear utility.

14.2 Supplemental Information

14.2.1 SAS1 Input File

```
=sas1
tn68-1d-rad, calc 972-07, GE 7x7, 40,000 MWD/MTU, 3.3wt%, 10 year
27N-18COUPLE  INFHOMMEDIUM
'Fuel-Basket Zone - without channel
uo2        1  den=1.885  1.0 293.  92235 3.3  92238 96.7 end
zircalloy   1  den=0.430  end
ss304       1  den=0.707  end
al          1  den=0.237  end
'Plenum-Basket Zone - without channel
zircalloy   2  den=0.359  end
ss304       2  den=0.614  end
al          2  den=0.185  end
'Top Fitting Zone - No Basket - without channel
zircalloy   3  den=0.182  end
ss304       3  den=0.309  end
'Bottom Fitting-Basket Zone - without channel
zircalloy   4  den=0.205  end
ss304       4  den=1.475  end
al          4  den=0.238  end
'Basket outer shell
ss304       5  1.0  end
'basket outer shell and shims and rails
al          6  1.0  end
'Cask Body, Outer Shell, Polydisc shells
carbonsteel  7  1.0  end
'Polypropylene disk
arbmpropylene 0.90  2  1  0  0  1001 14.3  6012 85.7 8 1.0 end
'Resin/Aluminum
arbmtnres   1.58  5  1  0  0  1001 5.05 5000 1.05 6012 35.13
8016 41.73  13027 14.93  9  0.904  end
al          9  0.096  end
end comp
end
last
tn68 gamma and neutron dose - 1 dimensionsal analysis - radial
cylindrical
1  50.0    75   -1  0.   0.  754.3  1.161E10
1  83.92   75   -1  0.   0.  754.3  1.161E10
5  84.4    1    0
6  88.27   6    0
7  107.32  27   0
9  122.56  19   0
7  124.47  3    0
end zone
0.01843 0.20989 0.23295 0.13102 0.17703 0.19296
0.03777 28z  0.00369 0.02404 0.02607 0.48033
0.04253 0.00998 0.01624 0.05803 0.07415 0.26490
0.01843 0.20989 0.23295 0.13102 0.17703 0.19296
0.03777 28z  0.00369 0.02404 0.02607 0.48033
0.04253 0.00998 0.01624 0.05803 0.07415 0.26490
read xsdose
365.76
end
```

14.3 References

1. Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors."

TABLE 14.1-1

DATA FOR TN68 ACTIVATION ANALYSIS

Component	Flux (n/cm^2/ s)	Composition	Element	% wt
Body & Lid	2.34E5	SA350LF3 and/or SA203	Mn	0.9
			Ni	3.75
			Fe	94.8
			C	0.2
			Si	0.35
Gamma Shield	2.34E5	SA105	Mn	1.0
			Fe	98.5
			C	0.2
			Si	0.3
Outer Shell	2.28E2	SA516 Gr 55	Mn	0.7
			Fe	99.3
Neutron Shield	4.63E3	Polyester Resin Mixture	H	5.05
			B	1.05
			C	35.13
			O	41.73
			Al	14.93
			Zn	2.11
Fuel Basket (poison assumed as Al)	4.91E5	SA240 (SS304)	Mn	2.0
			Cr	19.0
			Ni	9.5
			Si	0.75
			Fe	68.75
		SB-209 (Al 6061)	Si	0.6
			Mg	1.0
			Cr	0.2
			Cu	0.3
			Al	97.9

TABLE 14.1-2

RESULTS OF ORIGEN2 ACTIVATION ANALYSIS
Curies per Cask

Nuclide	Basket	Body, Lid and Rails	Resin and Al Boxes	Outer Shell & Protective Cover	Total
Cr^{51}	7.216E-3	6.625E-5	1.027E-7	5.660E-9	7.282E-03
Mn^{54}	6.527E-4	3.148E-3	-----	3.093E-7	3.801E-03
Fe^{55}	9.690E-3	4.640E-2	-----	4.557E-6	5.609E-02
Fe^{59}	1.789E-4	8.627E-4	-----	8.476E-8	1.042E-03
Co^{58}	8.856E-4	2.488E-4	-----	-----	1.134E-03
Co^{60}	1.235E-5	3.486E-6	4.636E-10	-----	1.584E-05
Ni^{63}	1.292E-3	3.630E-4	1.554E-09	-----	1.655E-03
Zn^{65}	-----	-----	6.926E-6	-----	6.926E-06
Ni^{59}	1.068E-5	3.001E-6	-----	-----	1.368E-05
H^{3}	-----	-----	1.618E-10	-----	1.618E-10
C^{14}	-----	2.066E-10	5.898E-10	-----	7.964E-10
TOTAL					7.104E-02

Note: Only the nuclides with activity greater than 10^{-10} curies and those listed in 10 CFR 61.55 are reported here.

TABLE 14.1-3

COMPARISON OF TN-68 ACTIVITY WITH CLASS A WASTE LIMITS

Specific Activity of Long-Lived Isotopes (10CFR61.55 Table 1)

Nuclide	Ci/m^3	Limit (Ci/m^3)	Volume (m^3)	Component
C^{14}	-----	80	1.72	Basket
Ni^{59}	6.21E-6	220		
C^{14}	3.24E-11	80	6.38	Body
Ni^{59}	1.67E-6	220		
C^{14}	1.33E-10	80	4.43	Resin

Specific Activity of Short-Lived Isotopes (10CFR61.55 Table 2)

Nuclide	Ci/m^3 "A"	Limit (Ci/m^3) "B"	Volume (m^3)	Component
Co^{60}	7.18E-6	700	1.72 m^3	Basket
Ni^{63}	7.51E-4	35		
$T_{1/2}<5$	1.08E-2*	700		
Co^{60}	5.46E-7	700	6.38 m^3	Body
Ni^{63}	5.69E-5	35		
$T_{1/2}<5$	7.95E-3*	700		
$T_{1/2}<5$	6.98E-6*	700	0.71 m^3	Shell
H^3	3.65E-11	40	4.43 m^3	Resin
Co^{60}	1.05E-10	700		
Ni^{63}	3.51E-10	35		
$T_{1/2}<5$	1.59E-6*	700		

* - Sum of isotopes with half-life less than 5 years (Cr51,Mn54,Fe55,Fe59,Co58,Zn65)